January 10, 1990

Docket No. 50-318

Mr. G. C. Creel
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
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Dear Mr. Creel:

SUBJECT: ISSUANCE OF AMENDMENT (UNIT 2 TACS 66456, 72037 AND 72975)

The Commission has issued the enclosed Amendment No. 123 to Facility Operating License No. DPR-69 for the Calvert Cliffs Power Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated February 12, 1988 and February 7, 1989, as supplemented on March 30, April 21, April 25 and May 8, 1989.

This amendment modifies the Unit 2 Technical Specifications and License to support Cycle 9 operations.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Scott Alexander McNeil

Scott Alexander McNeil, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 123 to DPR-69
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDI-1	PDI-1	OGC	PDI-1			
NAME	CVogan	SMcNeil/bah		RCapra			
DATE	11/14/89	11/15/89	11/29/89	01/10/90			

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PDR ADOCK 05000318
P PDC

Mr. G. C. Creel
Baltimore Gas & Electric Company

Calvert Cliffs Nuclear Power Plant

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Baltimore Gas and Electric Company (the licensee) dated February 12, 1988 and February 7, 1989, as supplemented on March 30, April 21 and April 25 and May 8, 1989, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-69 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 123 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 10, 1990

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 123 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-11	2-11
2-12	2-12
2-13	2-13
B2-1	B2-1
B2-2*	B2-2*
B2-3	B2-3
B2-4*	B2-4*
B2-5	B2-5
B2-6	B2-6
3/4 1-1	3/4 1-1
3/4 1-2*	3/4 1-2*
3/4 1-2a	3/4 1-2a
3/4 1-3*	3/4 1-3*
3/4 1-4*	3/4 1-4*
3/4 1-5*	3/4 1-5*
3/4 1-5a*	3/4 1-5a*
3/4 1-6*	3/4 1-6*
3/4 1-9*	3/4 1-9*
3/4 1-10*	3/4 1-10*
3/4 1-11*	3/4 1-11*
3/4 1-12*	3/4 1-12*
3/4 1-17*	3/4 1-17*
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 1-19A*	3/4 1-19A*
3/4 1-19B	3/4 1-19B
3/4 1-20*	3/4 1-20*
3/4 1-23*	3/4 1-23*
3/4 1-24*	3/4 1-24*
3/4 1-27	3/4 1-27
3/4 2-3	3/4 2-3
3/4 2-4	3/4 2-4
3/4 2-4a	3/4 2-4A
3/4 2-5*	3/4 2-5*
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a*	3/4 2-7A*
3/4 2-8	3/4 2-8
3/4 2-8 (cont.)	3/4 2-8A
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-10a	3/4 2-10A
3/4 2-11	3/4 2-11
3/4 2-12*	3/4 2-12*
3/4 2-13	3/4 2-13
3/4 2-14	3/4 2-14

AMENDMENT NO. 123 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NO. 50-318

Revise Appendix A as follows: (CONTINUED)

Remove Pages	Insert Pages
B3/4 1-1	B3/4 1-1
B3/4 1-1a	B3/4 1-1A
B3/4 1-2*	B3/4 1-2*
B3/4 1-3*	B3/4 1-3*
B3/4 1-4	B3/4 1-4
B3/4 1-5*	B3/4 1-5*
B3/4 2-1*	B3/4 2-1*
B3/4 2-2	B3/4 2-2
B3/4 7-2a	B3/4 7-2A
5-3*	5-3*
5-4	5-4

* Overleaf pages provided for continuity purposes only.

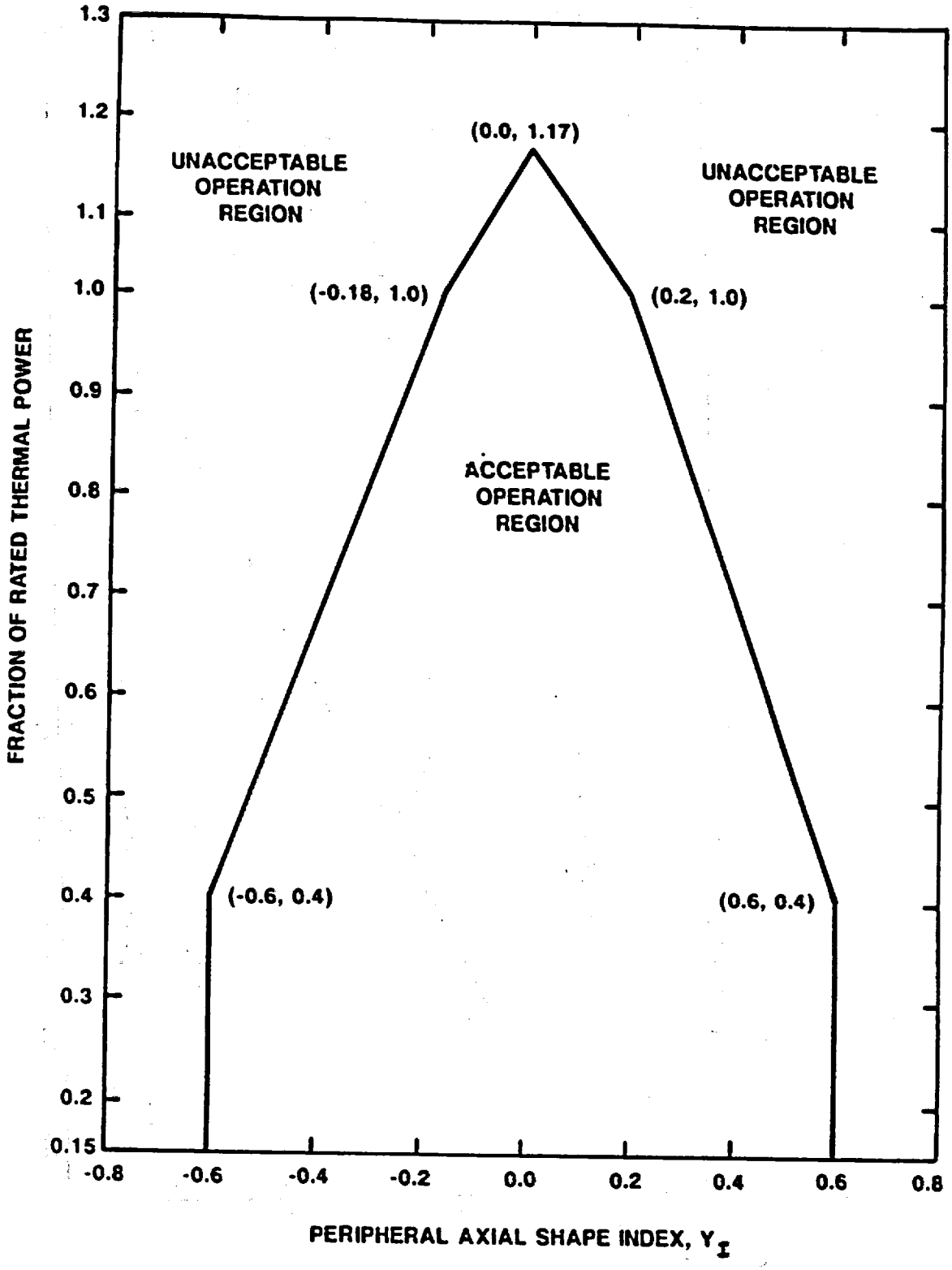

FIGURE 2.2-1
PERIPHERAL AXIAL SHAPE INDEX, Y_I
vs. FRACTION OF RATED THERMAL POWER

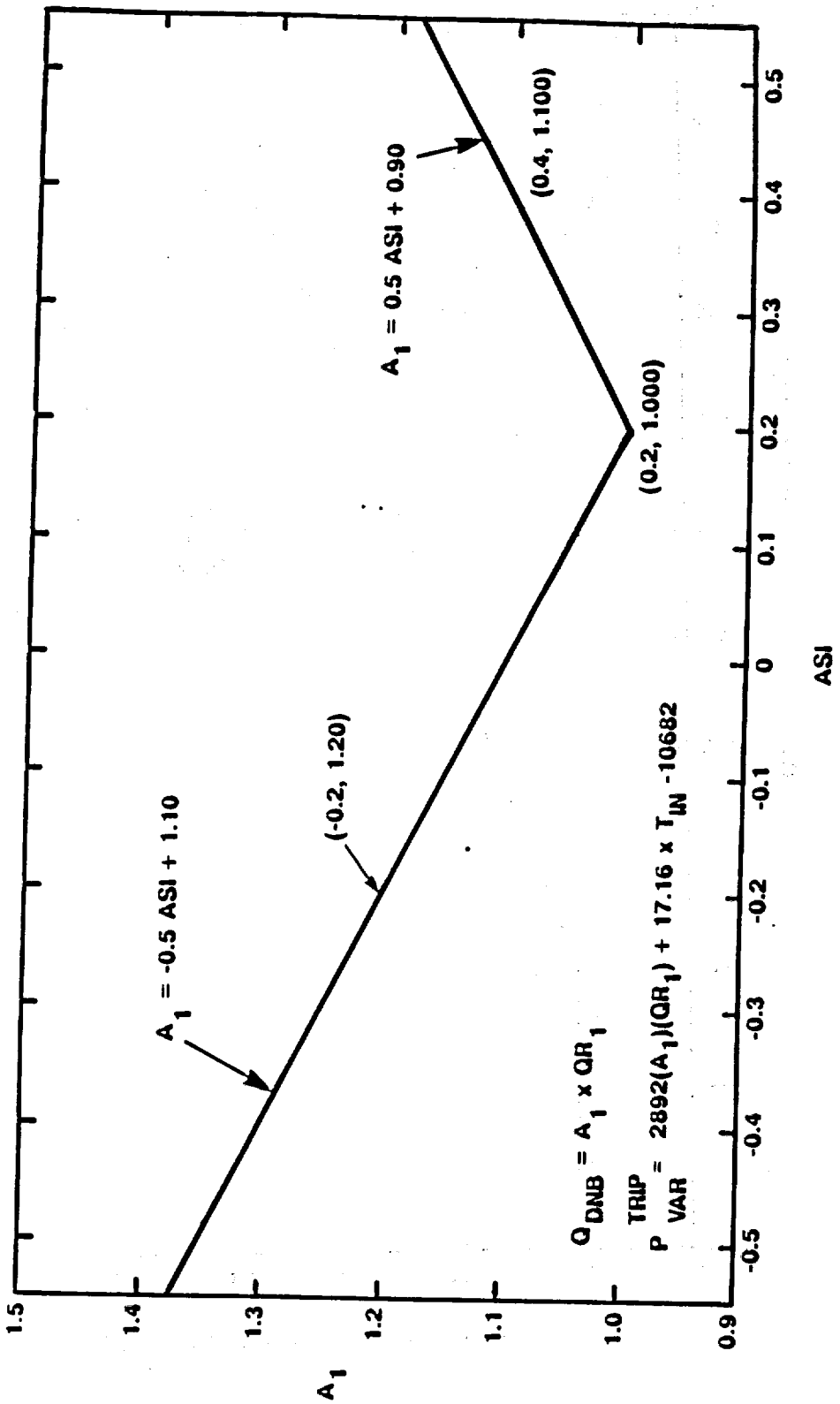

FIGURE 2.2-2
THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (ASI VERSUS A_1)

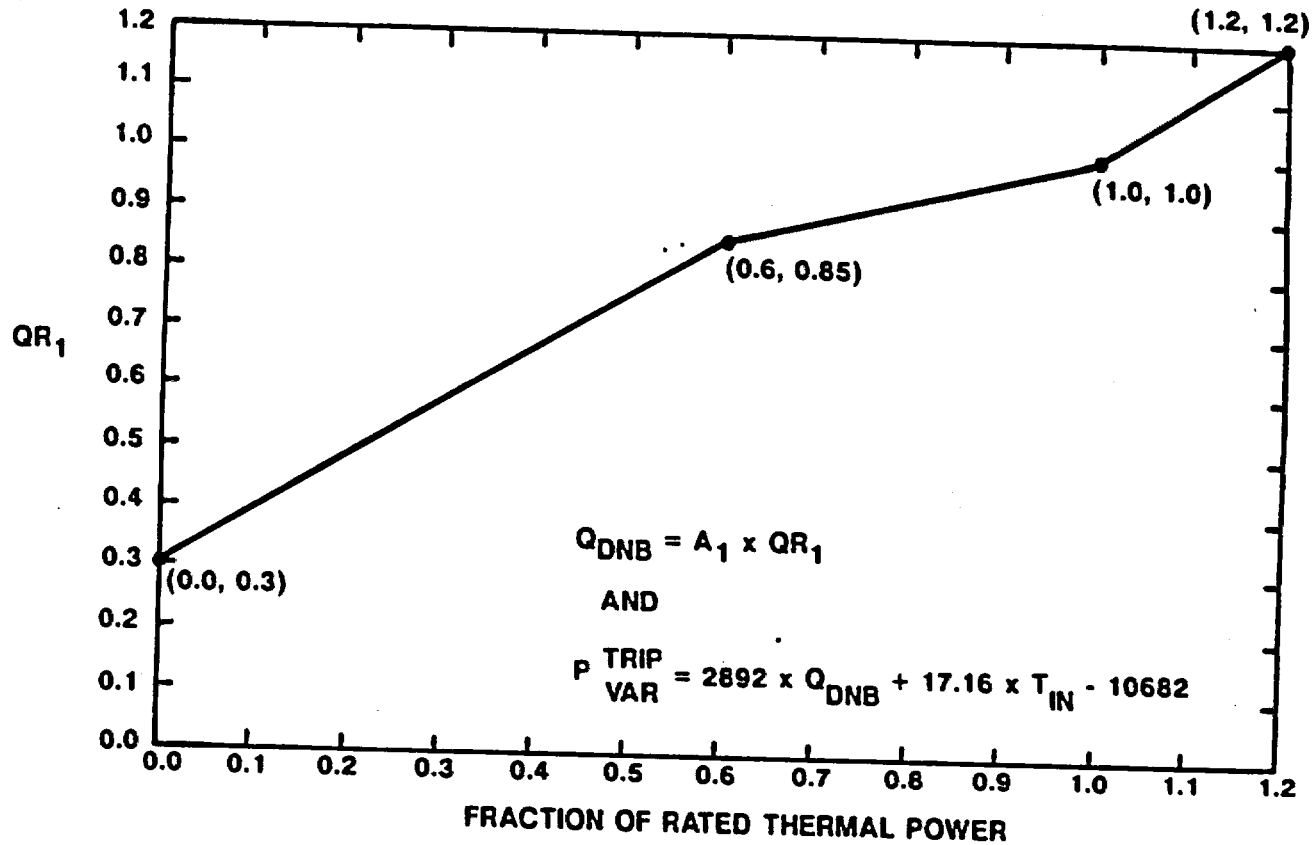

FIGURE 2.2-3

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which could result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 22.0 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore **THERMAL POWER** and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to the DNB SAFDL of 1.15 in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). This DNB SAFDL assures with at least a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curves of Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 show conservative loci of points of **THERMAL POWER**, Reactor Coolant System pressure and maximum cold leg temperature of various pump combinations for which the DNB SAFDL is not violated for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at **THERMAL POWER** levels higher than 110% of **RATED THERMAL POWER** is prohibited by the high power level trip setpoint specified in

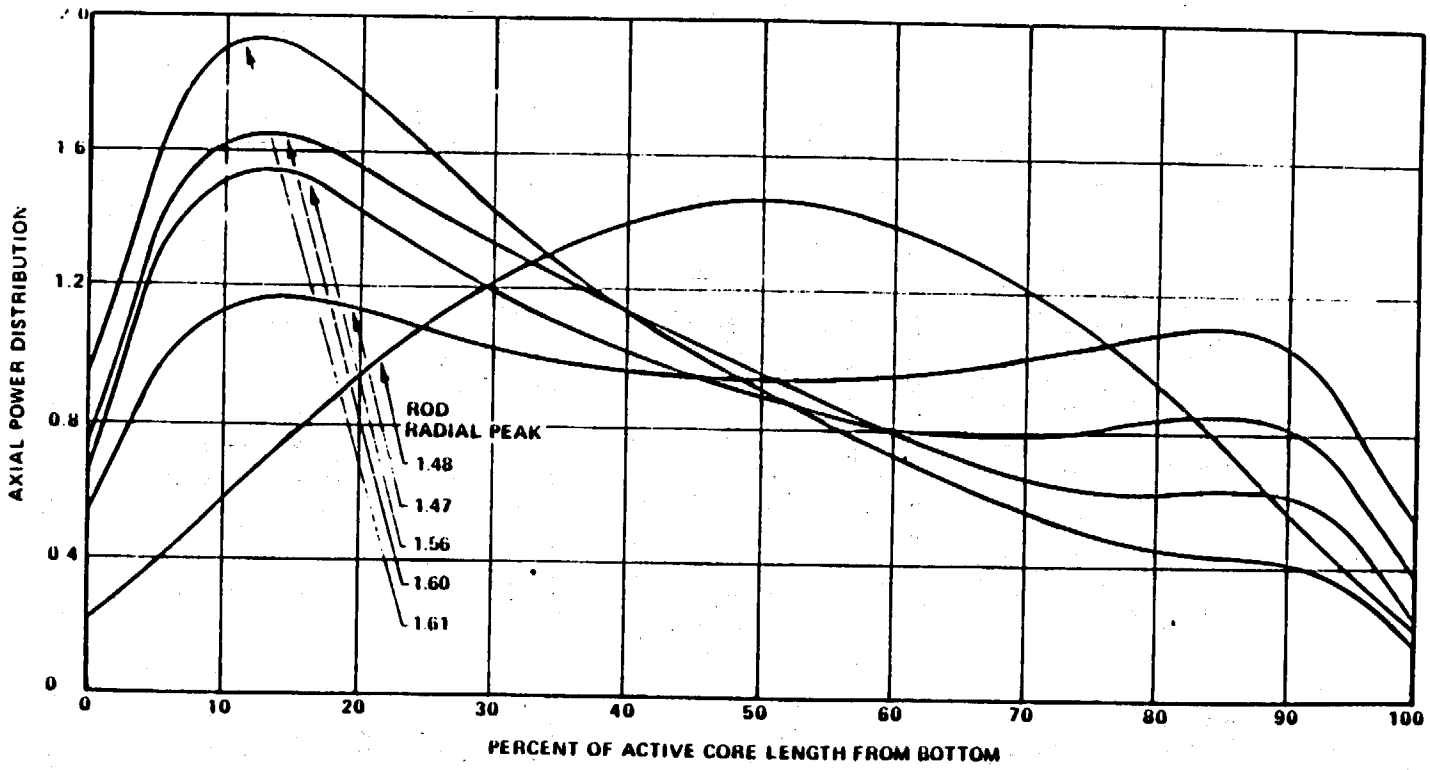

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

Table 2.1-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figures 2.1-1, 2.1-2, 2.1-3 and 2.1-4 to be valid are shown on the figures.

The reactor protective system, in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than 1.15, in conjunction with the ESCU methodology, and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from over pressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III, 1967 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves, and fittings are designed to ANSI B 31.7, Class I, 1969 Edition, which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 30% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 110% of RATED THERMAL POWER, which is the value used in the safety analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit

operation of the reactor at reduced power if one or two reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above the DNB SAFDL of 1.15, in conjunction with ESCU methodology, under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two- and three-pump operation prevents the minimum value of DNBR from going below the DNB SAFDL of 1.15, in conjunction with ESCU methodology, during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against over pressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to, or at least concurrently with, a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 685 psia is sufficiently below the full-load operating point of 850 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 85 psi in the accident analyses which was based on the Main Steam Line Break event.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the reactor coolant system will not exceed its Safety Limit. The specified setpoint in combination with the auxiliary feedwater actuation system ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event.

Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology, nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore-to-incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, and the number of reactor coolant pumps operating. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - Tavg > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The **SHUTDOWN MARGIN** shall be equal to or greater than the limit line of Figure 3.1-1b.

APPLICABILITY: MODES 1, 2**, 3, and 4.

ACTION:

With the **SHUTDOWN MARGIN** less than the limit line of Figure 3.1-1b, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or equivalent until the required **SHUTDOWN MARGIN** is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The **SHUTDOWN MARGIN** shall be determined to be equal to or greater than the limit line of Figure 3.1-1b:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required **SHUTDOWN MARGIN** shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% **RATED THERMAL POWER** after each fuel loading, by consideration of the factors of (e) below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

---

* Adherence to Technical Specification 3.1.3.6 as specified in Surveillance Requirements 4.1.1.1.1 assures that there is sufficient available shutdown margin to match the shutdown margin requirements of the safety analyses.

** See Special Test Exception 3.10.1.

\# With $K_{eff} \geq 1.0$.

\## With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

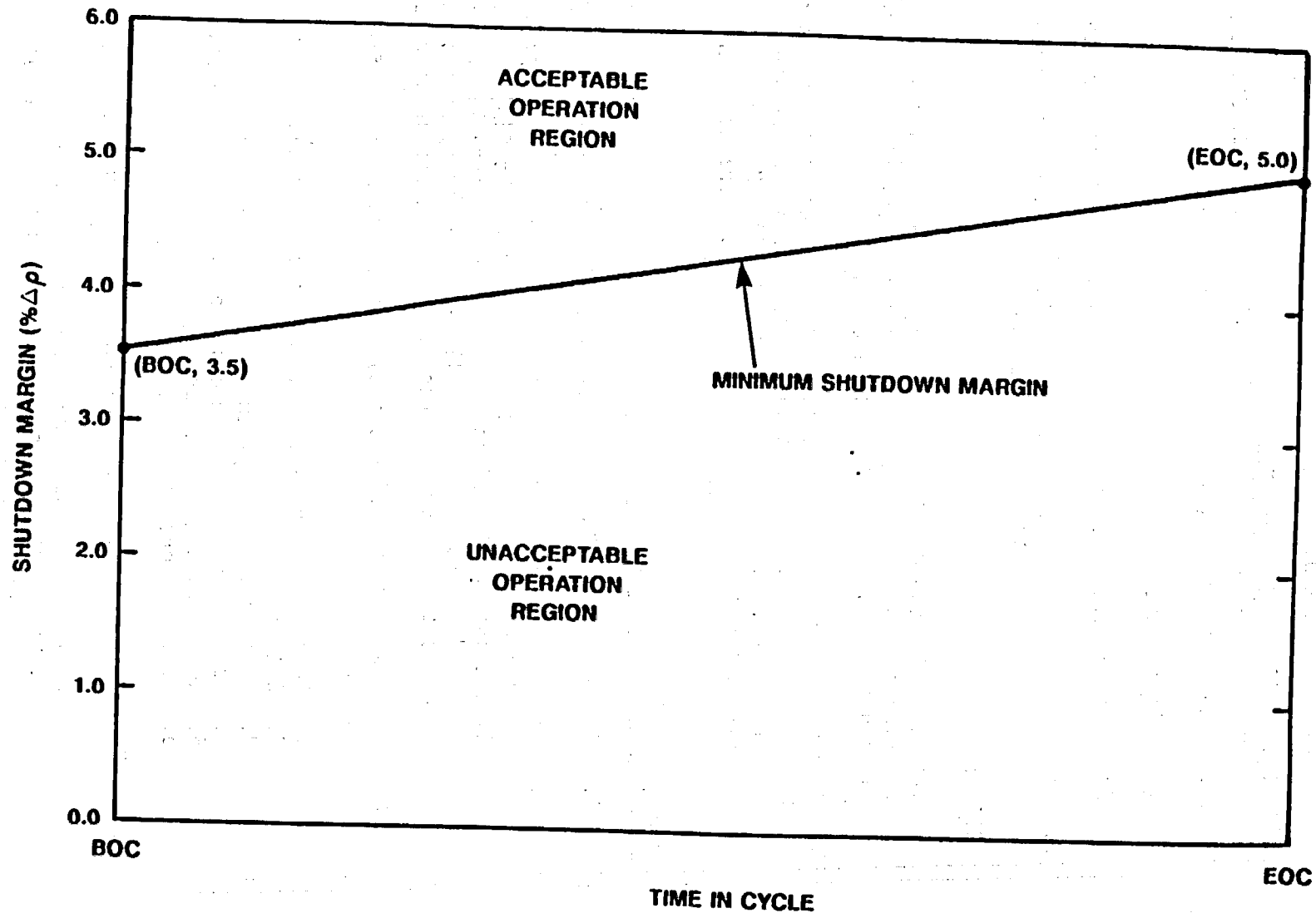
6.0
5.0
4.0
3.0
2.0
1.0
0.0
SHUTDOWN MARGIN (%Δρ)
ACCEPTABLE
OPERATION
REGION
(EOC, 5.0)
(BOC, 3.5)
MINIMUM SHUTDOWN MARGIN
UNACCEPTABLE
OPERATION
REGION
BOC
EOC
TIME IN CYCLE

FIGURE 3.1-1b

SHUTDOWN MARGIN - $T_{avg} \leq 200^{o}F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 3.0% $\Delta k/k$.

APPLICABILITY: MODE 5

a. Pressurizer level $\geq$ 90 inches from bottom of the pressurizer.

b. Pressurizer level < 90 inches from bottom of the pressurizer and all sources of non-borated water $\leq$ 88 gpm.

ACTION:

a. With the SHUTDOWN MARGIN < 3.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 2300 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

b. With the pressurizer drained to $\leq$ 90 inches and all sources of non-borated water > 88 gpm, immediately suspend all operations involving positive reactivity changes while the SHUTDOWN MARGIN is increased to compensate for the additional sources of non-borated water or reduce the sources of non-borated water to $\leq$ 88 gpm.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.0% $\Delta k/k$:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2. With the pressurizer drained to $\leq$ 90 inches determine:

a. Within one hour and every 12 hours thereafter that the level in the reactor coolant system is above the bottom of the hot leg nozzles, and

b. Within one hour and every 12 hours thereafter that the sources of non-borated water are $\leq$ 88 gpm or the shutdown margin has compensated for the additional sources.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be ≥ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm through the reactor coolant system.

REACTIVITY CONTROL SYSTEM

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than the limit line of Figure 3.1-1a, and

b. Less negative than -2.7×10^{-4} Δ k/k/°F at **RATED THERMAL POWER.**

APPLICABILITY: **MODES** 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least **HOT STANDBY** within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

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* With $K_{eff} \geq 1.0$.

\# See Special Test Exception 3.10.2.

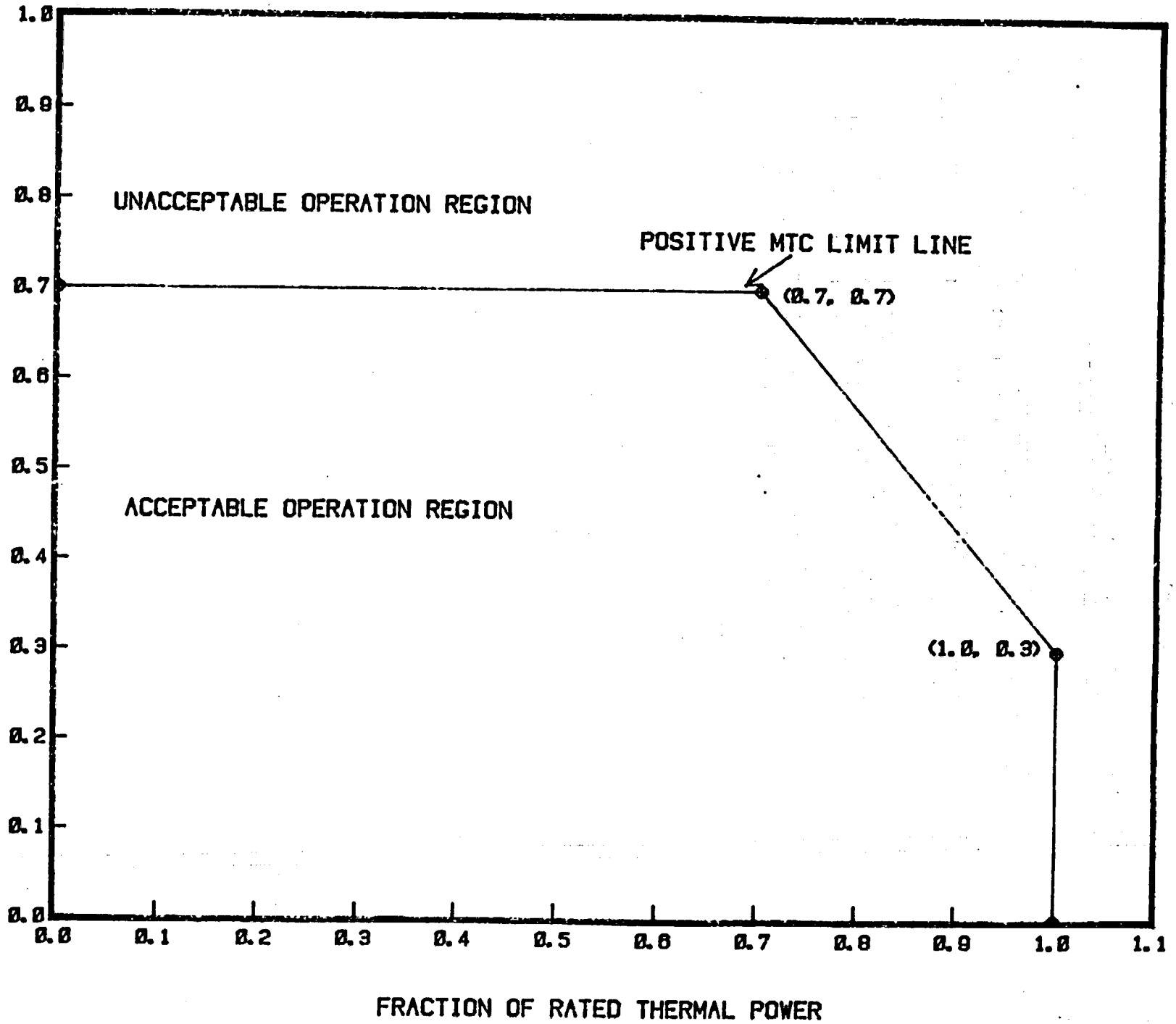
UNACCEPTABLE OPERATION REGION
POSITIVE MTC LIMIT LINE
(0.7, 0.7)
ACCEPTABLE OPERATION REGION
(1.0, 0.3)
ALLOWABLE POSITIVE MTC LIMIT ($10^{-4}\Delta\rho/^{0}F$)
FRACTION OF RATED THERMAL POWER
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1

Figure 3.1.1

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER above 90% of RATED THERMAL POWER, within 7 EFPD after initially reaching an equilibrium condition at or above 90% of RATED THERMAL POWER after each fuel loading.

c. At any THERMAL Power, within 7 EFPD of reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be **OPERABLE**:

a. Two flow paths from the boric acid storage tanks required to be **OPERABLE** pursuant to Specifications 3.1.2.8 and 3.1.2.9 via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System **OPERABLE**, restore at least two boron injection flow paths to the Reactor Coolant System to **OPERABLE** status within 72 hours or be in at least **HOT STANDBY** and borated to a **SHUTDOWN MARGIN** equivalent to at least 3% Δ k/k at 200°F within the next 6 hours; restore at least two flow paths to **OPERABLE** status within the next 7 days or be in **COLD SHUTDOWN** within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated **OPERABLE**:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the concentrated boric acid tanks is above the temperature limit line shown on Figure 3.1-1.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per refueling interval by verifying on a SIAS test signal that:

(1) each automatic valve in the flow path actuates to its correct position, and

(2) each boric acid pump starts.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be **OPERABLE.***

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump **OPERABLE**, restore at least two charging pumps to **OPERABLE** status within 72 hours or be in at least **HOT STANDBY** and borated to a **SHUTDOWN MARGIN** equivalent to at least 3% Δ k/k at 200°F within the next 6 hours; restore at least two charging pumps to **OPERABLE** status within the next 7 days or be in **COLD SHUTDOWN** within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated **OPERABLE**:

a. At least once per refueling interval by verifying that each charging pump starts automatically upon receipt of a Safety Injection Actuation Test Signal.

b. No additional Surveillance Requirements other than those required by Specification 4.0.5.

---

* Above 80% **RATED THERMAL POWER** the two **OPERABLE** charging pumps shall have independent power supplies.

BORIC ACID PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Motion Inhibit and all shutdown and regulating CEAs shall be **OPERABLE** with each CEA of a given group positioned within 7.5 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: **MODES 1* and 2***

ACTION:

a. With one or more CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, be in at least **HOT STANDBY** within 6 hours.

b. With the CEA Motion Inhibit inoperable, within 6 hours either:

   1. Restore the CEA Motion Inhibit to **OPERABLE** status, or

   2. Place and maintain the CEA drive system mode switch in either the "Off" or any "Manual Mode" position and fully withdraw all CEAs in groups 3 and 4 and withdraw the CEAs in group 5 to less than 5% insertion, or

   3. Be in at least **HOT STANDBY.**

c. With one CEA inoperable due to causes other than addressed by ACTION a, above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in **MODES** 1 and 2 may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year.

d. With one CEA inoperable due to causes other than addressed by ACTION a, above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in CEA group 5, operation in **MODES** 1 and 2 may continue.

---

* See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION

e. With one or more CEAs misaligned from any other CEAs in its group by more than 7.5 inches but less than 15 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided all of the following conditions are met:

a. The THERMAL POWER level shall be reduced to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination within one hour; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used.

b. Within one hour after reducing the THERMAL POWER as required by (a) above, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

f. With one CEA misaligned from any other CEA in its group by 15 inches or more, operation in MODES 1 and 2 may continue, provided that the misaligned CEA is positioned within 7.5 inches of the other CEAs in its group in accordance with the time allowance determined by the Better Axial Shape Selection System (BASSS) or, if the BASSS time allowance is unavailable, the time allowance shown in Figure 3.1-3. If Figure 3.1-3 is used, the pre-misaligned F_I value used to determine the allowable time to realign the CEA from Figure 3.1-3 shall be the latest measurement taken within 5 days prior to the CEA misalignment. If no measurements were taken within 5 days prior to the misalignment, a pre-misaligned F_I of 1.70 shall be assumed.

g. With one CEA misaligned from any other CEA in its group by 15 inches or more at the conclusion of the permitted time allowance, immediately start to implement the following actions:

1. If the THERMAL POWER level prior to the misalignment was greater than 50% of RATED THERMAL POWER, THERMAL POWER shall be reduced to less than the greater of:

a) 50% of RATED THERMAL POWER

b) 75% of the THERMAL POWER level prior to the misalignment

within one hour after exceeding the permitted time allowance.

2. If the THERMAL POWER level prior to the misalignment was ≤ 50% of RATED THERMAL POWER, maintain THERMAL POWER no higher than the value prior to the misalignment.

If negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Within one hour after establishing the appropriate THERMAL POWER as required above, either:

1. Restore the CEA to within the above specified alignment requirements, or

2. Declare the CEA inoperable. After declaring the CEA inoperable, POWER OPERATION may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided the remainder of the CEAs in the group with the inoperable CEA are aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

h. With more than one CEA inoperable or misaligned from any other CEA in its group by 15 inches (indicated position) or more, be in at least HOT STANDBY within 6 hours.

i. For the purposes of performing the CEA operability test of TS 4.1.3.1.2, if the CEA has an inoperable position indication channel, the alternate indication system (pulse counter or voltage dividing network) will be used to monitor position. If a direct position indication (full out reed switch or voltage dividing network) cannot be restored within ten minutes from the commencement of CEA motion, or CEA withdrawal exceeds the surveillance testing insertion by > 7.5 inches, the position of the CEA shall be assumed to have been > 15 inches from its group at the commencement of CEA motion.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Motion Inhibit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each CEA not fully inserted shall be determined to be **OPERABLE** by inserting it at least 7.5 inches at least once per 31 days.

4.1.3.1.3 The CEA Motion Inhibit shall be demonstrated **OPERABLE** at least once per 31 days by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit also prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

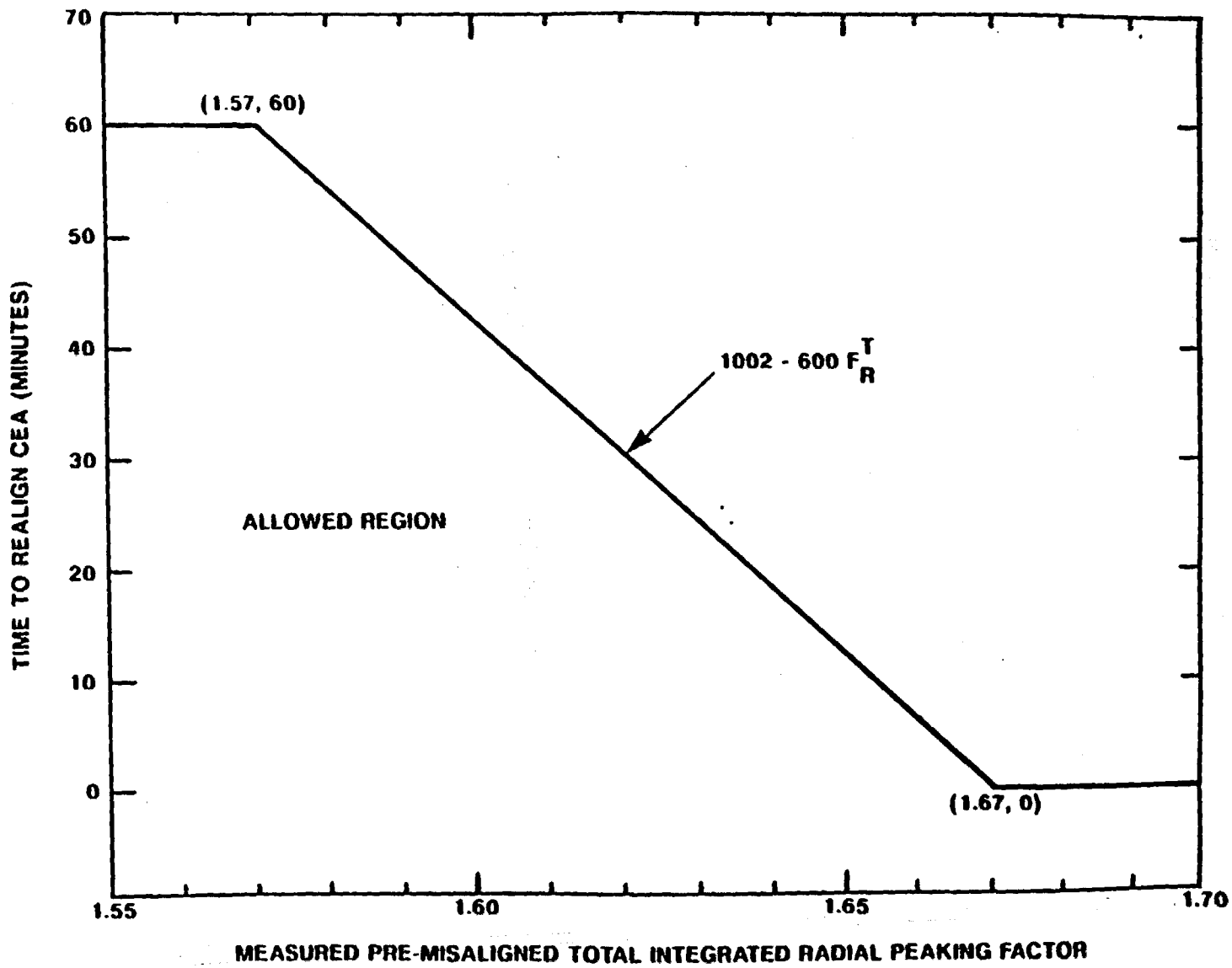

FIGURE 3.1-3

ALLOWABLE TIME TO REALIGN CEA VERSUS
INITIAL TOTAL INTEGRATED RADIAL PEAKING FACTOR

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CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515^{o}F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per refueling interval.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, within one hour either:

a. Withdraw the CEA to at least 129.0 inches, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With $K_{eff} \geq 1.0$

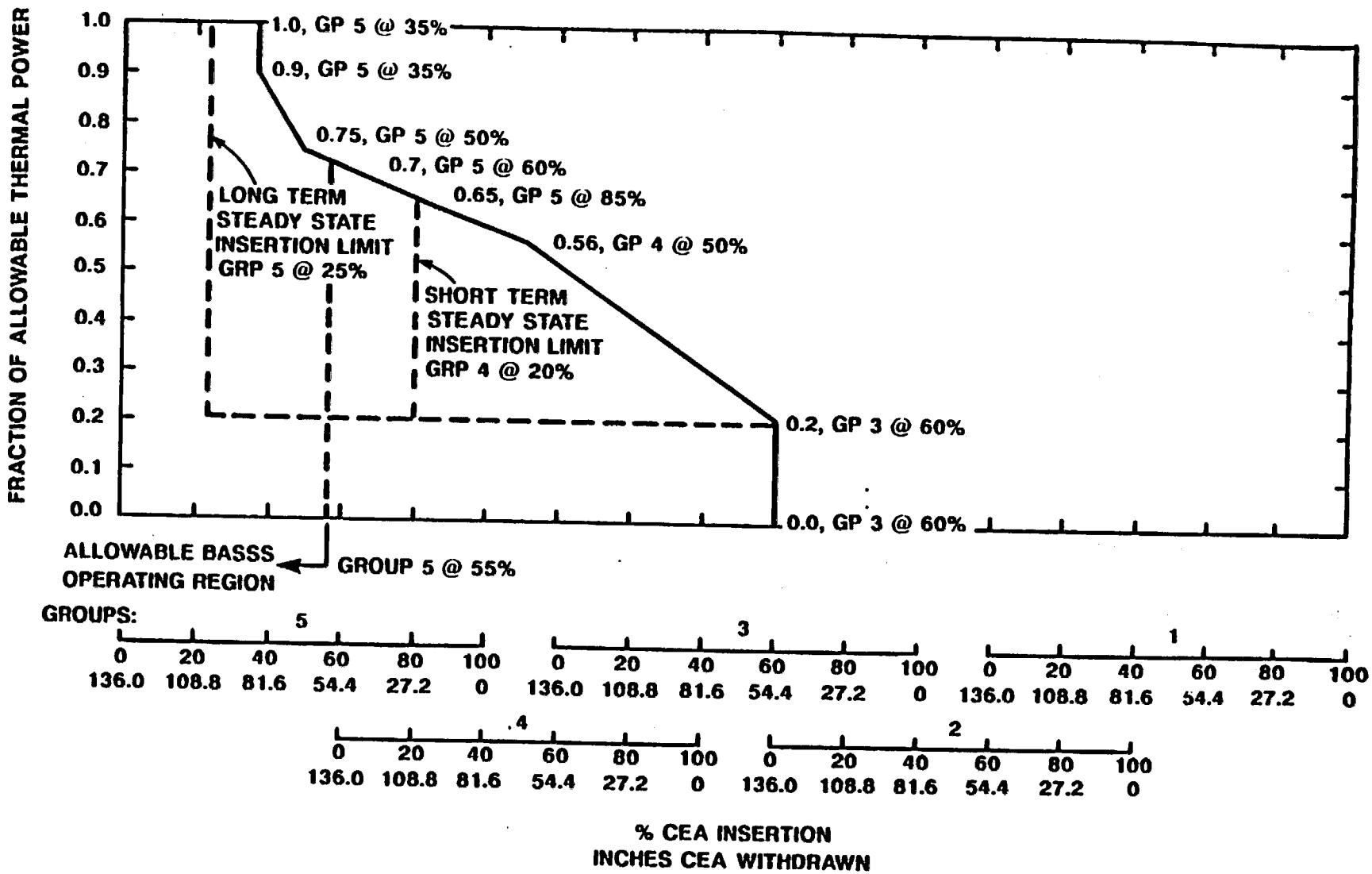


FIGURE 3.1-2

CEA GROUP INSERTION LIMITS vs. FRACTION OF ALLOWABLE THERMAL POWER
FOR EXISTING RCP COMBINATION

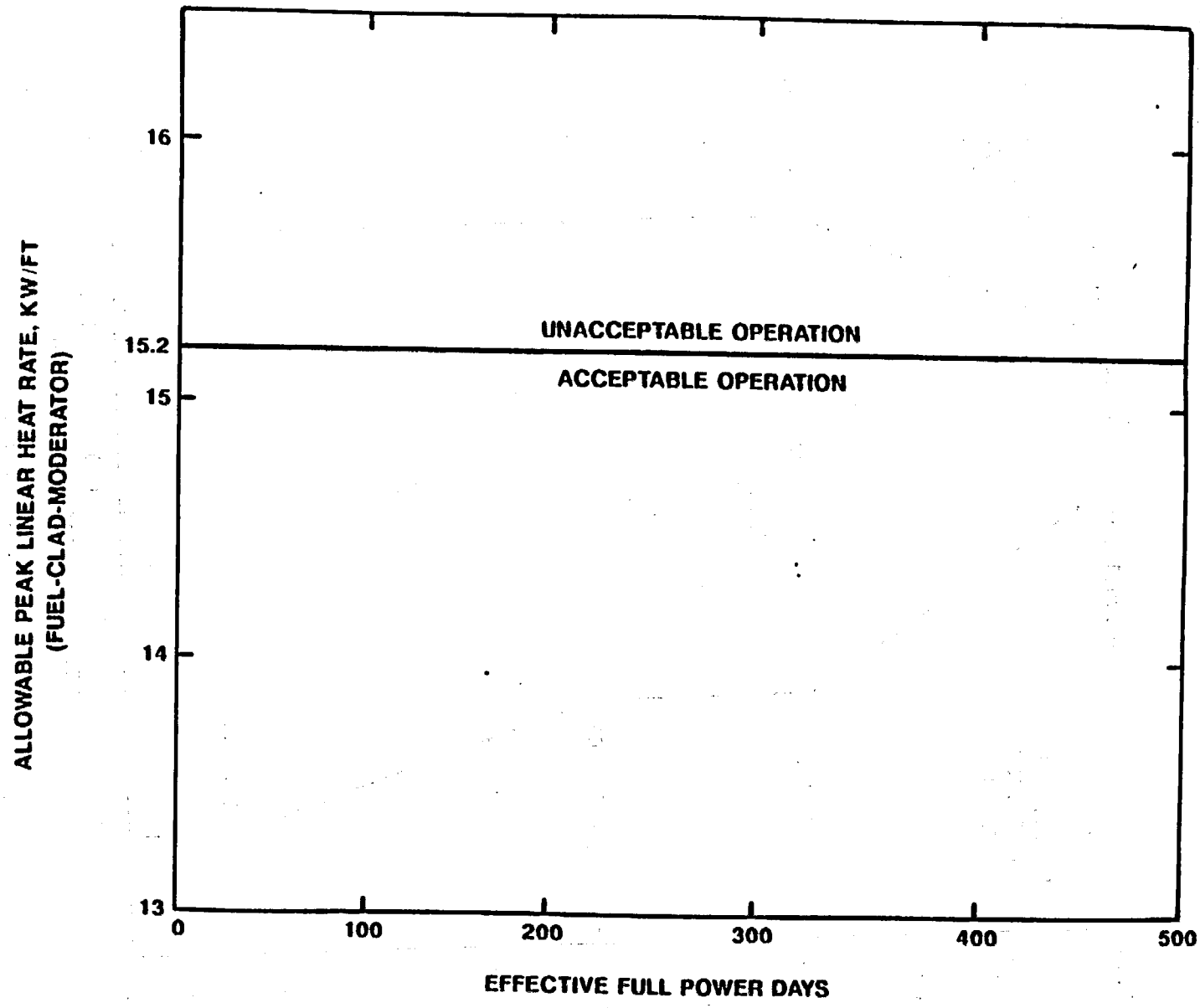


FIGURE 3.2-1

ALLOWABLE PEAK LINEAR HEAT RATE vs. BURNUP

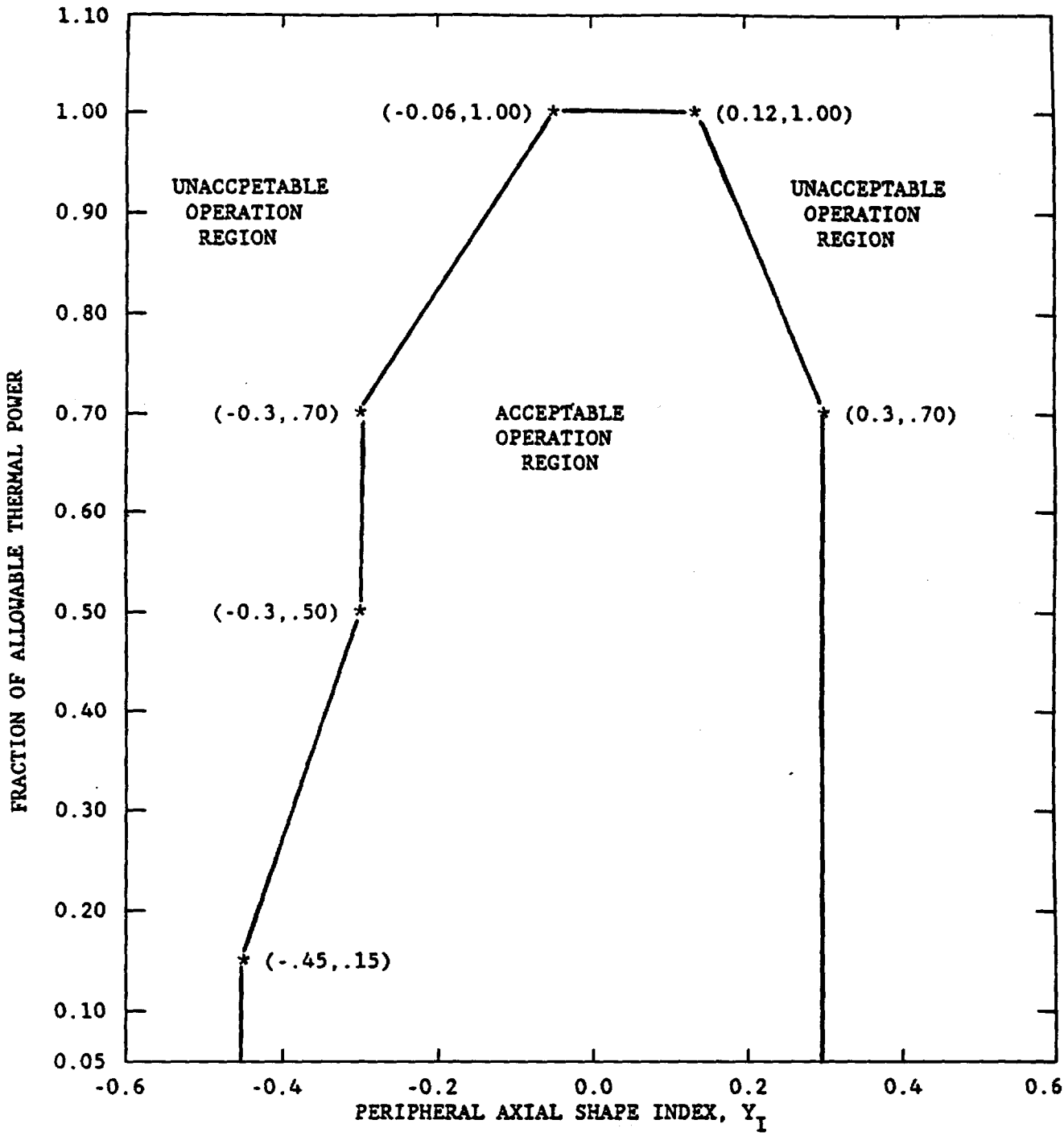


Figure 3.2-2
Linear Heat Rate Axial Flux Offset Control Limits

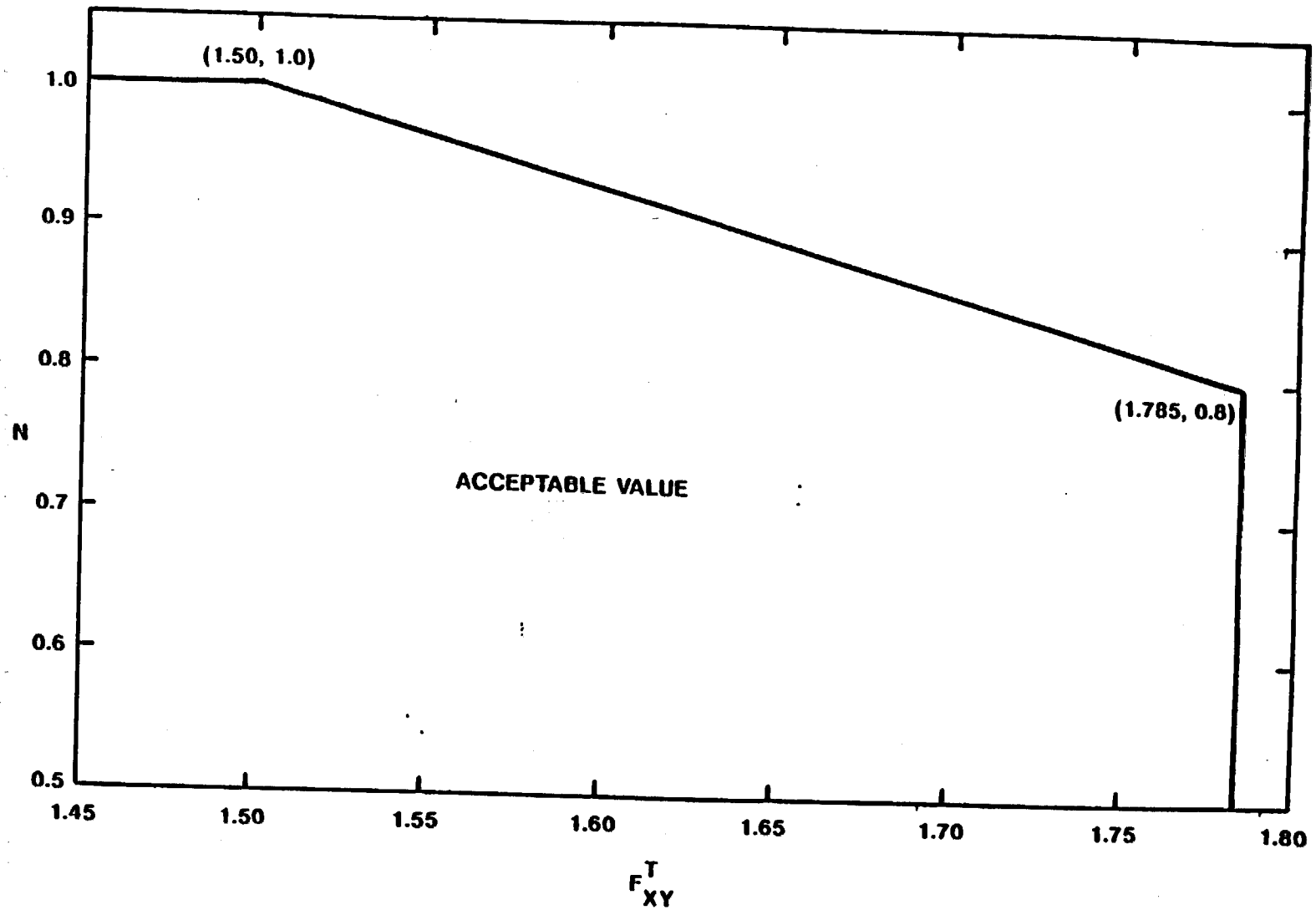


FIGURE 3.2-3b

TOTAL PLANAR RADIAL PEAKING FACTOR vs. N

DELETED

POWER DISTRIBUTION LIMITS

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2.1 The calculated value of F^T_{xy} shall be limited to $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.70, within 6 hours either:

a. Withdraw and maintain full length CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce THERMAL POWER as follows:

   1. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} within the limits of Figure 3.2-3a, or

   2. Reduce THERMAL POWER to less than or equal to the limit established by the Better Axial Shape Selection System (BASSS) as a function of F^T_{xy}; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+T_q)$ when F_{xy} is determined with a non-full core power distribution mapping system and shall be calculated as $F^T_{xy} = F_{xy}$ when determined with a full core power distribution mapping system. F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.030.

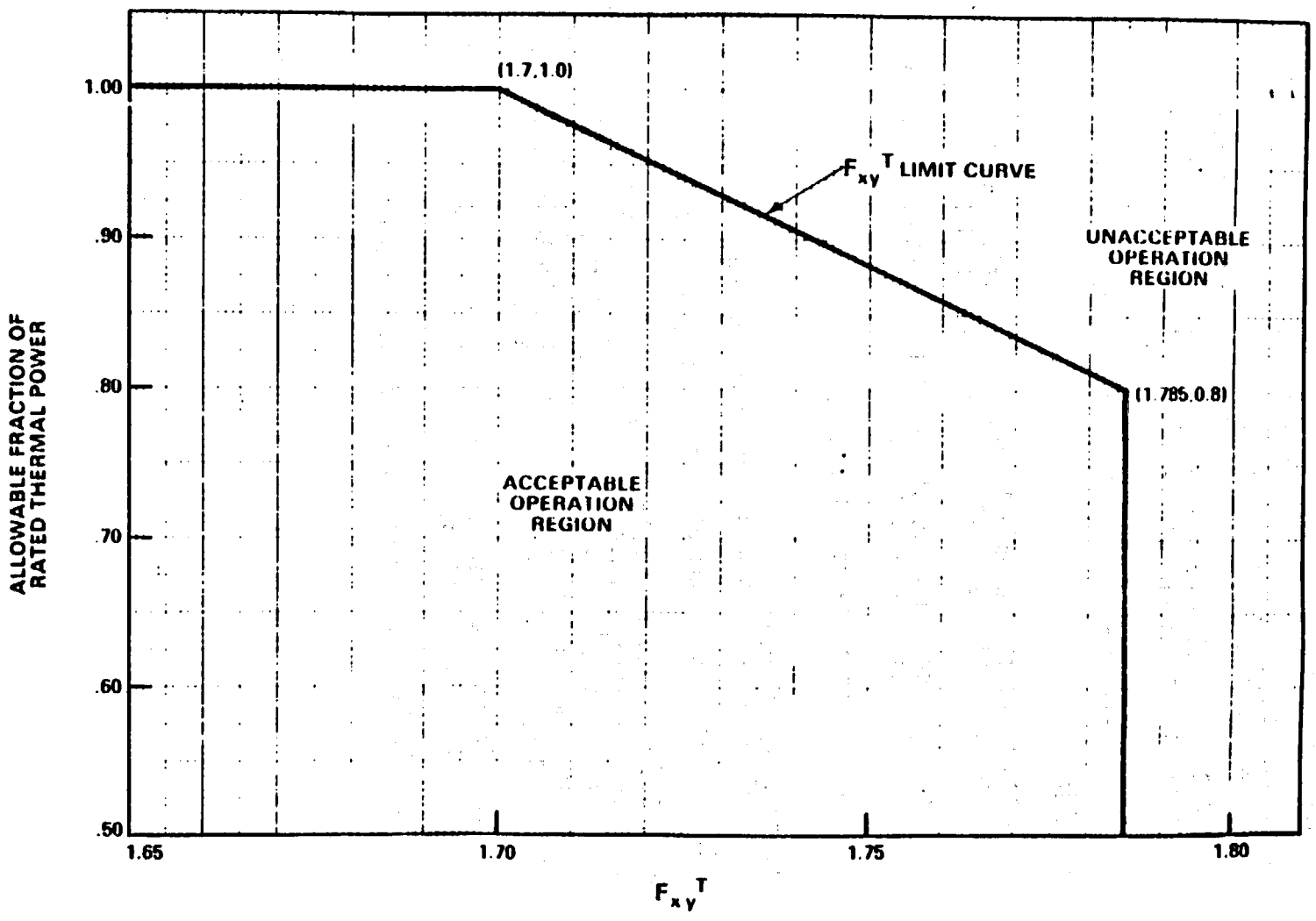
---

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.1.3 F_{xy}^{T} shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.1.4 T_q shall be determined each time a calculation of F_{xy}^{T} is made using a non-full core power distribution mapping system and the value of T_q used to determine F_{xy}^{T} shall be the measured value of T_q.



TOTAL PLANAR RADIAL PEAKING FACTOR vs ALLOWABLE FRACTION OF RATED THERMAL POWER

Figure 3.2-3a

POWER DISTRIBUTION LIMITS

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2.2 The value of N presently used in Specification 4.2.1.3 shall be in accordance with Figure 3.2-3b.

APPLICABILITY: MODE 1 when operating in accordance with Specification 4.2.1.3.

ACTION:

With the value of N presently used in Specification 4.2.1.3 exceeding the limit shown in Figure 3.2-3b, within 6 hours either:

a. Reduce the value of N used in Specification 4.2.1.3 to within the limits of Figure 3.2-3b; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}\ (1+T_q)$ when F_{xy} is determined with a non-full core power distribution mapping system and shall be calculated as $F^T_{xy} = F_{xy}$ when determined with a full core power distribution mapping system. N shall be determined to be within its limit by monitoring F^T_{xy} at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 3 days of accumulated operation in MODE 1.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.2.3 F^T_{xy} shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

4.2.2.2.4 T_q shall be determined each time a calculation of F^T_{xy} is made using a non-full core power distribution mapping system and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be limited to $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.70, within 6 hours either:

a. Be in at least **HOT STANDBY**, or

b. Withdraw and maintain the full length CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce **THERMAL POWER** as follows:

1. Reduce **THERMAL POWER** to bring the combination of **THERMAL POWER** and F_r^T within the limits of Figure 3.2-3c, or

2. Reduce **THERMAL POWER** to less than or equal to the limit established by the Better Axial Shape Selection System (BASSS) as a function of F_r^T.

When the **THERMAL POWER** is determined from Figure 3.2-3c, it shall be used to establish a revised upper **THERMAL POWER LEVEL** limit on Figure 3.2-4 (i.e., Figure 3.2-4 shall be truncated at the allowable fraction of **RATED THERMAL POWER** determined by Figure 3.2-3c). Subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r\ (1+T_q)$ when F_r is determined with a non-full core power distribution mapping system and shall be calculated as $F_r^T = F_r$ when determined with a full core power distribution mapping system. F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of **RATED THERMAL POWER** after each fuel loading,

b. At least once per 31 days of accumulated operation in **MODE** 1, and

c. Within four hours if the **AZIMUTHAL POWER TILT** (T_q) is > 0.030.

*See Special Test Exception 3.10.2.

4.2.3.3 F_r^T shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is made using a non-full core power distribution mapping system and the value of T_q used to determine F_r^T shall be the measured value of T_q.

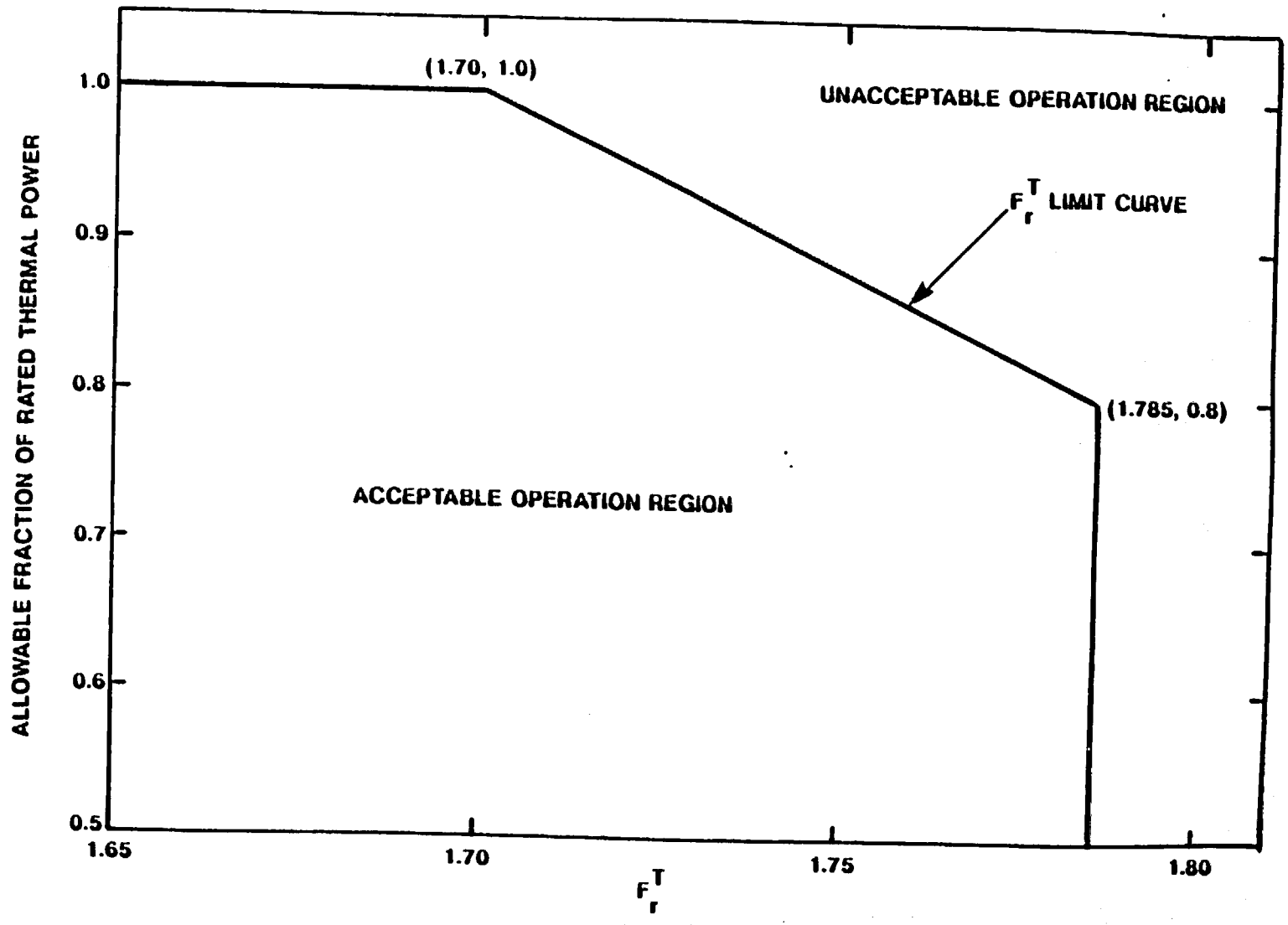


FIGURE 3.2-3c

TOTAL INTEGRATED RADIAL PEAKING FACTOR vs. ALLOWABLE FRACTION OF RATED THERMAL POWER

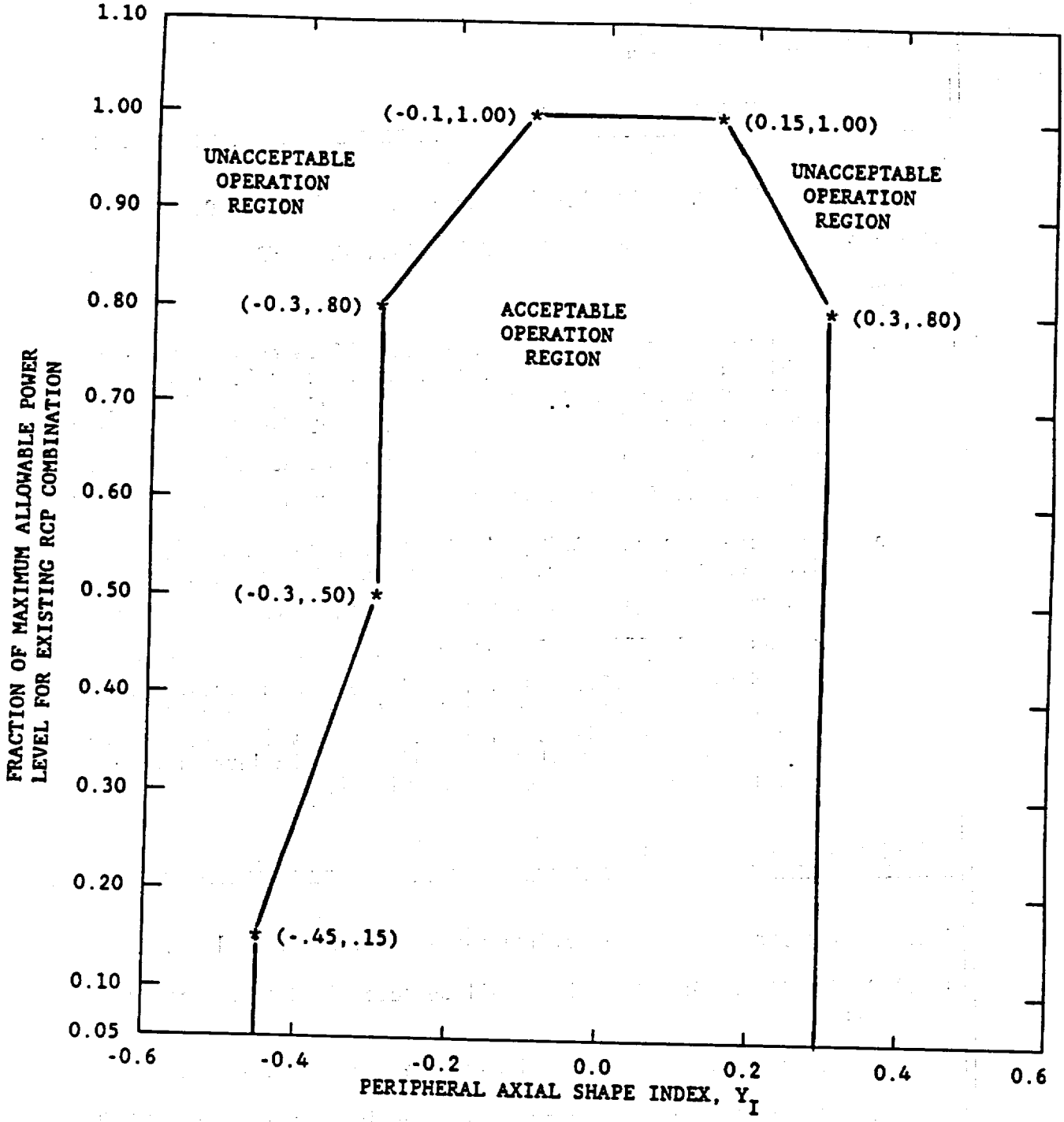


Figure 3.2-4
DNB Axial Flux Offset Control Limits

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.030.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 12 hours, and

b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX, THERMAL POWER

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

4.2.5.3 The Better Axial Shape Selection System (BASSS) may be used for monitoring THERMAL POWER as a function of AXIAL SHAPE INDEX. BASSS monitoring shall be limited to CEA insertions of the lead bank ≤ 55%.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING	THREE REACTOR COOLANT PUMPS OPERATING	TWO REACTOR COOLANT PUMPS OPERATING - SAME LOOP	TWO REACTOR COOLANT PUMPS OPERATING-OPPOSITE LO
Cold Leg Temperature	≤ 548° F	**	**	**
Pressurizer Pressure	≥ 2200 psia*	**	**	**
Reactor Coolant System Total Flow Rate	≥ 370,000 gpm	**	**	**
AXIAL SHAPE INDEX, THERMAL POWER	***	**	**	**

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

** These values left blank pending NRC approval of ECCS analyses for operation with less than four reactor coolant pumps operating.

*** The AXIAL SHAPE INDEX, THERMAL POWER shall be maintained within the limits established by the Better Axial Shape Selection System (BASSS) for CEA insertions of the lead bank of ≤ 55% when BASSS is OPERABLE, or within the limits of FIGURE 3.2-4 for CEA insertions specified by FIGURE 3.1-2.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient **SHUTDOWN MARGIN** ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting **SHUTDOWN MARGIN** requirement at beginning of cycle is determined by the requirements of several transients, including Boron Dilution and Steam Line Rupture. The **SHUTDOWN MARGIN** requirements for these transients are relatively small and nearly the same. However, the most limiting **SHUTDOWN MARGIN** requirement at end of cycle comes from just one transient, the Steam Line Rupture event. The requirement for this transient at end of cycle is significantly larger than that for any other event at that time in cycle and, also, considerably larger than the most limiting requirement at beginning of cycle.

The variation in the most limiting requirement with time in cycle has been incorporated into Technical Specification 3.1.1.1, in the form of a specified **SHUTDOWN MARGIN** value which varies linearly from beginning to end of cycle. This variation in specified **SHUTDOWN MARGIN** is conservative relative to the actual variation in the most limiting requirement. Consequently, adherence to Technical Specification 3.1.1.1 provides assurance that the available **SHUTDOWN MARGIN** at any time in cycle will exceed the most limiting **SHUTDOWN MARGIN** requirement at that time in cycle.

In **MODE 5**, the reactivity transients resulting from any event are minimal and do not vary significantly during the cycle. Therefore, the specified **SHUTDOWN MARGIN** in **MODE 5** via Technical Specification 3.1.1.2 has been set equal to a constant value which is determined by the requirement of the most limiting event at any time during the cycle, i.e., Boron Dilution with the pressurizer level less than 90 inches and the sources of non-borated water restricted. Consequently, adherence to Technical Specification 3.1.1.2 provides assurance that the available **SHUTDOWN MARGIN** will exceed the most limiting **SHUTDOWN MARGIN** requirement at any time in cycle.

3/4.1.1 BORATION CONTROL

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The system also provides coolant flow following a SIAS (e.g., during a Small Break LOCA) to supplement flow from the Safety Injection System. The Small Break LOCA analyses assume flow from a single charging pump, accounting for measurement uncertainties and flow maldistribution effects in calculating a conservative value of charging flow actually delivered to the RCS. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 3.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 6500 gallons of 7.25% boric acid solution from the boric acid tanks or 55,627 gallons of 2300 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during MODES 1, 2, 3 and 4. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 3% Δk/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 737 gallons of 7.25% boric acid solution from the boric acid tanks or 9,844 gallons of 2300 ppm borated water from the refueling water tank.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA and to a large misalignment (≥ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (≤ 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment (≥ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. The **ACTION** statement associated with a large CEA misalignment requires prompt action to realign the CEA to avoid excessive margin degradation. If the CEA is not realigned within the given time constraints, action is specified which will preserve margin, including reductions in **THERMAL POWER**.

For a single CEA misalignment, the time allowance to realign the CEA (Figure 3.1-3 or as determined by BASSS) is permitted for the following reasons:

1. The margin calculations which support the power distribution LCOs for DNBR are based on a steady-state F_r^I as specified in Technical Specification 3.2.3.

2. When the actual F_r^T is less than the Technical Specification value, additional margin exists.

3. This additional margin can be credited to offset the increase in F_r^I with time that will occur following a CEA misalignment due to xenon redistribution.

The requirement to reduce power level after the time limit of Figure 3.1-3 or the time limit determined by BASSS is reached offsets the continuing increase in F_r^I that can occur due to xenon redistribution. A power reduction is not required below 50% power. Below 50% power there is sufficient conservatism in the DNB power distribution LCOs to completely offset any, or any additional, xenon redistribution effects.

The **ACTION** statements applicable to misaligned or inoperable CEAs include requirements to align the **OPERABLE** CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors, and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

BASES

Operability of the CEA position indicators is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the OPERABILITY and the ACTION statements applicable to inoperable CEA position indicators permit continued operations when positions of CEAs with inoperable indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The surveillance requirements affecting CEAs with inoperable position indication channels allow 10 minutes for testing each affected CEA. This time limit was selected so that 1) the time would be long enough for the required testing, and 2) if all position indication were lost during testing, the time would be short enough to allow a power reduction to 70% of maximum allowable thermal power within one hour from when the testing was initiated. The time limit ensures CEA misalignments occurring during CEA testing are corrected within the time requirements required by existing specifications.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the accident analyses. Measurements with $T_{avg} \geq 515^{o}$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The LSSS setpoints and the power distribution LCOs were generated based upon a core burnup which would be achieved with the core operating in an essentially unrodded configuration. Therefore, the CEA insertion limit specifications require that during MODES 1 and 2, the full length CEAs be nearly fully withdrawn. The amount of CEA insertion permitted by the Steady State Insertion Limits of Specification 3.1.3.6 will not have a significant effect upon the unrodded burnup assumption but will still provide sufficient reactivity control. The Transient Insertion Limits of Specification 3.1.3.6 are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels; however, long term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAP INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 3) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) a measurement-calculational uncertainty factor of 1.062, 2) an engineering uncertainty factor of 1.03, 3) an allowance of 1.002 for axial fuel densification and thermal expansion, and 4) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy} or F^T_r or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure

and Local Power Density - High LCOs and LSSS setpoints remain valid. An **AZIMUTHAL POWER TILT** > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F_{xy}^T = F_{xy}(1 + T_q)$ and $F_r^T = F_r(1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}^T, F_r^T and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of **RATED THERMAL POWER** provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNB SAFDL of 1.15 throughout each analyzed transient.

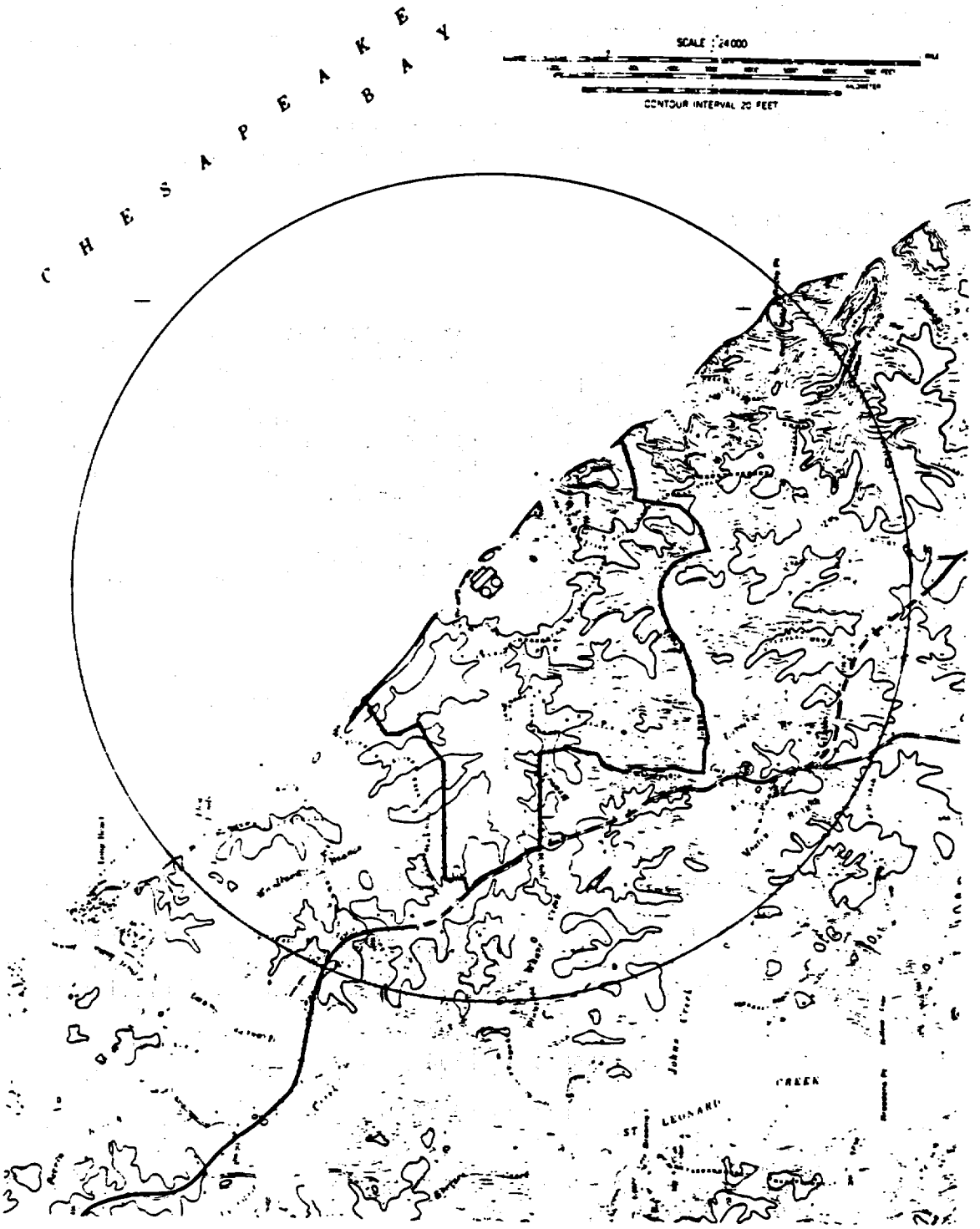
In addition to the DNB criterion, there are two other criteria which set the specification in Figure 3.2-4. The second criterion is to ensure that the existing core power distribution at full power is less severe than the power distribution factored into the small-break LOCA analysis. This results in a limitation on the allowed negative **AXIAL SHAPE INDEX** value at full power. The third criterion is to maintain limitations on peak linear heat rate at low power levels resulting from Anticipated Operational Occurrences (AOOs). Figure 3.2-4 is used to assure the LHR criterion for this condition because the linear heat rate LCO, for both ex-core and in-core monitoring, is set to maintain only the LOCA kw/ft requirements which are limiting at high power levels. At reduced power levels, the kw/ft requirements of certain AOOs (e.g., CEA withdrawal), tend to become more limiting than that for LOCA.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.7.1.2 (Continued)

(3)	Main Steam Line Break	1550 gpm Auxiliary Feedwater Flow (this being the maximum flow through the AFW suction line, with one unit requiring flow, prior to pump cavitation due to low NPSH).

At 10 minutes after an Auxiliary Feedwater Actuation Signal, the operator is assumed to be available to increase or decrease auxiliary feedwater flow to that required by the existing plant condition.



LOW POPULATION ZONE

FIG. 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 50 psig and a temperature of 276°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 3000 grams uranium. The initial core loading shall have a maximum enrichment of 2.99 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.35 weight percent U-235.

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemblies shall be sleeved with a sleeve design previously approved by the NRC.

CONTROL ELEMENT ASSEMBLIES

5.3.3 The reactor core shall contain 77 full length and no part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer, which is 700°F.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-69

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-318

INTRODUCTION

By its letters dated February 12, 1988 and February 7, 1989, as supplemented on March 30, April 21, April 25 and May 8, 1989, the Baltimore Gas and Electric Company (BG&E; the licensee) submitted requests to amend its operating license and Technical Specifications (TS) for Calvert Cliffs Unit No. 2 to support operation for a ninth cycle at a 100% rated core power of 2700 MWt. Cycle 9 will have a 24 month cycle length as did the previous cycle. The Unit 1 Cycle 10 design is the reference design for Unit 2 Cycle 9.

The March 30, 1989 supplement provided additional information concerning core burnup expected for Cycle 9 while the April 21 and 25, 1989 letters added further discussion of the potential effects of intermediate core loading configurations with higher enrichment fuel. Finally, the May 8, 1989 submittal provided camera-ready copies of the proposed TS changes.

These supplemental submittals did not affect the proposed TS changes noticed in the Federal Register (54 FR 14714) on April 12, 1989 and did not affect the staff's proposed determination that no significant hazards would result from these changes.

The NRC staff has reviewed the applications and the supporting documents and has prepared the following evaluation of the fuel design, nuclear design, thermal-hydraulic design, transient and accident analysis, and TS changes.

2.0 EVALUATION OF FUEL DESIGN

2.1 Fuel Assembly Description

The Cycle 9 core consists of 217 fuel assemblies. Ninety-two fresh (unirradiated) Batch L assemblies manufactured by Combustion Engineering (CE) will replace previously irradiated assemblies. Four of these Batch L assemblies will contain erbia (Er_2O_3) as the burnable poison material instead of B_4C. These assemblies will have an initial enrichment of 3.81 weight percent U-234 and are being placed in the Cycle 9 core as lead test assemblies to aid in qualification of the use of erbia as an acceptable burnable poison for use in 24-month cycle cores.

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2.2 Mechanical Design

The mechanical design of the CE Batch L reload fuel is essentially identical to the Batch M fuel approved for use in Cycle 10 of Calvert Cliffs Unit 1 with the following exceptions:

(a) The fuel rod plenum spring in the Batch L fuel has been redesigned to maximize the available rod internal void volume. This modification helps reduce high end of cycle (EOC) internal gas pressures.

(b) The overall length of the Batch L B_4C burnable poison rod has been increased so that the poison rod length is now the same as the fuel rod length. This allows the same type cladding tube to be used for both rod types.

(c) The size and number of crimp holes in the upper end of each of the five guide tubes of each Batch L assembly have been modified. This design change allows the fuel assembly upper end fitting guide tube posts to be reusable if the assembly must be disassembled for fuel rod reconstitution.

(d) The Batch L lower end fitting, flow hole configuration has been modified to a new smaller hole, more debris-resistant design. In this design, nine small, chamfered holes replace each of the larger holes in the reference cycle design, thus forming a smaller diameter flow path more restrictive to the intrusion of reactor coolant system debris into the fuel assembly.

The staff finds these design changes have been adequately considered in all aspects of the nuclear, mechanical, thermal-hydraulic, and transient safety analyses for Cycle 9. In addition, all CE fuel to be loaded for the Cycle 9 core was reviewed to ascertain that adequate shoulder gap clearance exists. Analyses were performed with NRC approved models and the licensee concluded that all shoulder gap and fuel assembly length clearances are adequate for Cycle 9. Thus, the changes are acceptable.

2.3 Thermal Design

The thermal performance of the CE fuel in Cycle 9 was evaluated using the FATES3B fuel evaluation model which has been approved by the NRC for BG&E licensing submittals. The licensee analyzed a composite standard fuel rod that enveloped the various fuel batches in Cycle 9. The analysis modeled the power and burnup levels representative of the peak rod at each burnup interval and bounds the erbia bearing fuel rods. Although the burnup range analyzed for the peak rod was greater than that expected at the end of Cycle 9, approximately one percent of the fuel rods will achieve burnups greater than the 52,000 MWD/T value approved for CE fuel if Cycles 8 and 9 are operated to their maximum burnups. In order to limit the maximum internal hot gas pressure throughout Cycle 9 to a value that is less than the nominal reactor coolant system (RCS) pressure of 2250 psia, the allowable peak linear heat

generation rate (PLHGR) in the peak rod was reduced to 15.2 Kw/ft. The licensee has confirmed that the maximum relative power density of any rod which exceeds 52,000 MWD/T will be at least 30% below the single rod peak in the core and the maximum pressure within any of these rods also will not reach the nominal RCS pressure.

Based on the above analysis performed using NRC approved methodology and on the fact that evaluations performed by the licensee have shown that the four erbia lead test assemblies are thermally compatible with the other fuel assemblies and meet all the appropriate fuel thermal design criteria required by the staff, the staff concludes that the thermal design of the Cycle 9 fuel is acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

3.1 Fuel Management

The Cycle 9 core consists of 217 fuel assemblies, each having a 14 by 14 fuel rod array. The highest U-235 enrichment occurs in the non-erbia bearing Batch L fuel assemblies which contain an assembly average enrichment of 4.30 weight percent U-235. The Calvert Cliffs fuel storage facilities have been analyzed for storage of fuel of maximum enrichment of 5.0 weight percent U-235 and, therefore, the fresh Batch L assemblies are acceptable from a fuel storage aspect.

The Calvert Cliffs refueling procedures allowed placement of fuel assemblies in intermediate positions during core alterations. During an analysis of refueling configurations, the licensee discovered that the potential existed for placing several 4.3 weight percent fresh highly reactive fuel assemblies together and losing shutdown margin below the minimum required 5 percent, or in the extreme, having an inadvertent criticality. Since a significant amount of reactivity could be added to a subcritical geometry by placing a single fresh 4.3 weight percent fuel assembly in certain intermediate locations, subscritical multiplication may no longer provide adequate warning of an approach to critically. Taking no credit for control element assemblies (CEAs) and assuming that the minimum refueling boron concentration exists, it was determined that an inadvertent criticality could occur under the extreme conditions of grouping a number of such highly reactive assemblies together. Therefore, BG&E issued a written report, in accordance with 10 CFR Part 21, "Reporting of Defects and Noncompliance," to the NRC on March 15, 1989, detailing the potential loss of shutdown margin that could occur during refueling.

As documented in a letter from George C. Creel (BG&E) to NRC dated April 21, 1989, BG&E has revised their refueling procedures to ensure that a fuel assembly will not be placed in an intermediate position during core alternations without first verifying its potential reactivity. Fuel will only be positioned in intermediate core locations which will contain fuel of equal or greater reactivity in the final core configuration. In order to prevent inadvertent misplacement of fuel assemblies, the revised procedures also require operators

to identify a fuel assembly as new or irradiated by its appearance. An irradiated assembly will have an oxide layer and appear black while new fuel will not. In addition, all core locations will be verified prior to insertion and after insertion. These additional procedures should ensure that a new fuel assemblies are placed where they are specified to be positioned. The staff finds this to be an acceptable means for assuring that the required shutdown margin is maintained at all times during refueling and that an inadvertent criticality will not occur.

The Cycle 9 core will use a low-leakage fuel management scheme. With the proposed loading, the Cycle 9 reactivity lifetime for full power operation is expected to be 20,650 MWD/T based on a Cycle 8 length of 18,300 MWD/T. The analyses presented by the licensee will accommodate a Cycle 9 length between 20,400 MWD/T and 21,500 MWD/T based on Cycle 8 lengths between 17,000 MWD/T and 19,000 MWD/T.

3.2 Power Distribution

Hot full power (HFP) fuel assembly relative power densities are provided in the reload analysis report for beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) unrodded confirguations. Radial power distributions at BOC and EOC are also provided for CEA Bank 5, the lead regulating bank, fully inserted. These distributions are characteristic of the high burnup end of the Cycle 8 shutdown window and tend to increase the radial power peaking in the Cycle 9 core. The four Batch LE lead test assemblies (Er_2O_3) were calculated to have maximum pin power peaking at least 10% lower than the maximum pin peaking in the core under all expected Cycle 9 operating conditions. The distributions were calculated with approved methods and include the increased power peaking that is characteristic of fuel rods adjacent to water holes. In addition, the safety and setpoint analyses conservatively include uncertainties and other allowances so that the power peaking values actually used are higher than those expected to occur at any time in Cycle 9. Therefore, the predicted Cycle 9 power distributions are acceptable.

3.3 Moderator Temperature Coefficient (MTC)

The Cycle 9 moderator temperature coefficient (MTC) positive limit varies linearly from $+0.3\times10^{-4}$ delta rho/°F at 100% power to $+0.7\times10^{-4}$ delta rho/°F at 70% power and below. The negative limit is -2.7×10^{-4} delta rho/°F. The NRC has previously expressed concern about positive MTC effects on the generic anticipated transients without scram (ATWS) assumptions and BG&E has stated that they will address the generic ATWS implications, if any, in the future. In the interim, the NRC has approved operation for core designs with allowable positive MTC values provided that the MTC becomes negative at 100% power and equilibrium xenon conditions. The licensee has agreed to this commitment and has predicted a negative MTC at HFP equilibrium conditions ranging from -0.04×10^{-4} delta rho/°F at BOC to -2.3×10^{-4} delta rho/°F at EOC for Cycle 9.

3.4 Control Requirements

The CEA worths and shutdown margin requirements for Unit 2 Cycle 9 are most limiting at EOC. The licensee's assessed shutdown margin requirements for Cycle 9 are based on the results of the EOC, hot zero power (HZP), steamline break event. After consideration of all reactivity uncertainties and biases, a worst case assessment for Cycle 9 results in a 0.4% delta rho margin in excess of the proposed 5.0% delta rho EOC TS requirement. Therefore, sufficient CEA worth is available to accommodate the reactivity effects of the steamline break event at the worst time during Cycle 9 allowing for the most reactive CEA stuck in the fully withdrawn position. Based on a review of the licensee's analysis, the staff finds the control requirements are satisfactorily met.

3.5 Incore Monitoring

The incore detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in the same manner as those for the reference cycle. However, changes to the on-line incore limiting condition for operation (LCO) monitoring system have been proposed for Cycle 9.

The INCA computer code, which is currently used for power distribution surveillance, would be replaced by the CECOR 3.3 code. Since CECOR 3.3 calculates a full core solution and can be used to obtain 3-dimensional power distribution data as compared to INCA which gives an octant solution, this change should result in more accurate monitoring. CECOR 3.3 would be used on-line and is virtually identical to and gives the same results as the CECOR 2.0 code which is currently utilized off-line. The use of CECOR 3.3 in an on-line network to monitor compliance with the linear heat rate (LHR) and departure from nucleate boiling (DNB) LCO TS would incorporate present separate monitoring systems, specifically the alarm Limit System for LHR and the Better Axial Shape Selection System (BASSS) for DNB, into one linked system. BASSS would use the CECOR 3.3 calculated radial peaking factor instead of the presently required TS limit. CECOR 3.3 would also supply BASSS with a core average axial shape index based on the full core solution which would be used together with the Bank 5 rod position and the unrodded radial peaking factor to calculate the available DNB overpower margin and alarms as at present. BASSS would provide the capability to monitor the LCO on total planar radial peaking factor, F_{xy}^{t} and total integrated radial peaking factor, F_{r}^{t}. If the TS limits on these were exceeded during normal operation, BASSS would activate an alarm and would calculate the proper trade-off with maximum allowed power to ensure that the axial power distribution and thermal margin/low pressure (TM/LP) trips remain conservative. The proposed TS are worded to support either the full core CECOR 3.3 measured power distribution or the present INCA octant measured power distributions.

The staff finds these proposed changes acceptable and appropriately included in the Cycle 9 safety analyses and proposed TS modifications.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

4.1 DNBR Analysis

Steady state Departure from Nucleate Boiling Ratio (DNBR) analyses of Cycle 9 at the rated power level of 2700 MWt have been performed using the approved core thermal-hydraulic code TORC and CETOP, and the CE-1 critical heat flux correlation. The cycle specific TORC and CETOP models used for designing Cycle 9 account for the small flow hole configuration used in the lower end fitting of the Batch L fresh fuel. Engineering hot channel factors and conservatisms are combined statistically with other uncertainty methods using the approved Extended Statistical Combination of Uncertainties (ESCU) methodology to arrive at an equivalent DNBR limit of 1.15 at a 95/95 probability/confidence level. The Cycle 9 DNBR analyses bound the four Batch LE lead test assemblies without crediting ESCU methods since the maximum single fuel rod peak is at least 10% below the maximum single fuel rod peak in the core. Therefore, the hottest fuel rod in the core is never located in any of the lead test assemblies at any point throughout Cycle 9. The DNBR analysis for Cycle 9 is, therefore, acceptable.

4.2 Fuel Rod Bowing

The fuel rod bow penalty accounts for the adverse impact on minimum DNBR of random variations in spacing between fuel rods. The methodology for determining rod bow penalties for Calvert Cliffs was based on NRC approved methods. The penalty at 45,000 MWD/T burnup is 0.006 in MDNBR. This penalty is included in the ESCU uncertainty allowance discussed above. For those assemblies with average burnup in excess of 45,000 MWD/T, sufficient margin exists to offset rod bow penalties. The staff, therefore, concludes that the analysis of fuel rod bow penalty is acceptable.

5.0 EVALUATION OF SAFETY ANALYSES

5.1 Non-LOCA Events

All of the key input parameters for the transient and accident analyses are bounded by (or are conservative with respect to) those of the reference cycle (Unit 1 Cycle 10) with the exception of the following:

1. Cycle 9 Batch L fuel utilities a small flow hole debris resistant design.

2. The maximum auxiliary feedwater (AFW) flow assumed in the safety analyses was increased from 1300 gpm for the reference cycle to 1550 gpm for Cycle 9 in anticipation of larger measured AFW flow.

3. The maximum assumed number of plugged U-tubes per steam generator was increased to 500 for all but the small break LOCA Cycle 9 analyses.

The licensee has determined by reanalysis or reevaluation that the results of all events affected by these input parameter changes remain bounded by those of the reference cycle. Based on a review of this analysis, the staff concludes that the non-LOCA transient and accident events for Cycle 9 are bounded by the reference analyses and the results of the non-LOCA safety analyses are acceptable.

5.2 LOCA Events

The large break LOCA has been reevaluated for Cycle 9 to demonstrate that a peak linear heat generation rate (PLHGR) of 15.5 Kw/ft complies with the acceptance criteria of 10 CFR 50.46 for emergency core cooling systems (ECCS) for light water reactors. Blowdown hydraulic calculations have shown that the slight increase in pressure drop due to the Batch L debris-resistant fuel design is insignificant. In addition, a reduction of 260 gpm in low pressure safety injection (LPSI) flow was assumed in order to establish an increased margin between analysis flow values and actual measured flow values. The licensee confirmed that there remains adequate safety injection flow to maintain a full downcomer with the reduced flow and, therefore, the reduction in LPSI flow does not affect the results. Since all other fuel related parameters for Cycle 9 remain bounded by those of the reference cycle, the staff concludes that the large break LOCA is bounded by the reference cycle analysis. Therefore, operation of Unit 2 Cycle 9 at a PLHGR of 15.5 Kw/ft and a power level of 2754 MWt (102% of 2700 MWT) is in compliance with the 10 CFR 50.46 acceptance criteria. The allowable PLHGR is being decreased to 15.2 Kw/ft for Cycle 9 to accommodate the more limiting fuel performance data associated with this second 24 month cycle in Unit 2.

The licensee reports that analyses have confirmed that small break loss of coolant accident (SBLOCA) results for Calvert Cliffs Unit 1 Cycle 8, which is the reference cycle for SBLOCA, bound the Unit 2 Cycle 9 results. Unlike the large break LOCA analysis which considered 500 plugged tubes per steam generator, the SBLOCA considered only 150 plugged tubes per steam generator. This effect as well as the effect of the reduction in LPSI flow and the incorporation of the debris-resistant fuel design were evaluated by the licensee and were found to have no significant impact on the SBLOCA. Therefore, acceptable SBLOCA ECCS performance is also demonstrated at a PLHGR of 15.5 Kw/ft and a reactor power level of 2754 MWt. The allowable PLHGR is being decreased to 15.2 Kw/ft for Cycle 9 to accommodate the more limiting fuel performance data associated with this second 24 month cycle in Unit 2. The staff has reviewed the licensee's analysis and concludes that it complies with 10 CFR 50.46 and the results are acceptable.

5.3 Fuel Handling Accident

The licensee evaluated the potential impact of an increase in core fuel enrichment to 4.35 weight percent U-235 and concluded that the design basis accidents (DBAs), previously analyzed by the licensee in the Calvert Cliffs Updated Final Safety Analysis Report (UFSAR) (Revision 8, August 24, 1988), bound any potential radiological consequences of DBAs that could result with 4.35 weight percent U-235.

The NRC staff reviewed the licensee's submittals as compared to NUREG/CR 5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors" and the Standard Review Plan for Pressurized Water Reactors. The staff agrees that the only potential DBA consequences that could result from the use of higher enrichment fuel, would be an increase in the thyroid doses that could result from a postulated design basis fuel handling accident. NUREG/CR 5009

estimates that the I-131 fuel gap actively in the peak fuel rod with 60,000 MWD/MT burnup (5.292 weight percent U-235) could be as high as 12%. This value is approximately 20% higher than the value normally used by the staff in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage facilities for Boiling and Pressurized Water Reactors").

The NRC staff, consequently, reevaluated the fuel handling accident for the Calvert Cliffs Unit 2 facility with an increase in iodine gap activity in the fuel damaged for a postulated fuel handling accident. The following table provides 1) handling accident thyroid doses presented in the NRC Safety Evaluation Report for Calvert Cliffs, dated August 28, 1972, and 2) the increased (by 20%) thyroid doses resulting from fuel initially enriched to 5.292 weight percent U-235, with burnup to 60,000 MWD/MT. The resulting doses are small fractions of the applicable regulatory requirements at 10 CFR Part 100.

Based upon the above, the staff concludes that 1) the bounding doses potentially increased are the thyroid doses at the Exclusion Area and Low Population Zone boundaries resulting from postulated fuel handling accidents, 2) these doses remain well within the 300 Rem thyroid exposure guideline values set forth in 10 CFR Part 100, and 3) the small calculated increase is not significant. Consequently, the staff has determined that the proposed increase in the core fuel enrichment limit to 4.35 weight percent U-235 is acceptable.

Table 1

Thyroid Doses as a Consequences of DBA Fuel Handling Accident

Exclusion Area		Low Population Zone	
Thyroid Dose (Rem)		Thyroid Dose (Rem)	
A*	B**	A*	B**
5.	6.	2.	2.4

*A Original SER dose (8/28/72)
**B Doses with higher enrichment and burnup (5.29%; 60,000 MWD/MT)

6.0 TECHNICAL SPECIFICATIONS

The following paragraphs summarize the proposed changes to the TS requested to support operation of Unit 2 Cycle 9. Some of these changes have already been implemented for Unit 1 Cycle 10. Others involve specific Unit 2 Cycle 9 changes such as the use of the new on-line incore LCO monitoring system discussed in Section 3.5 of this SER.

a. Thermal Margin/Low Pressure Trip Setpoint

TS Figures 2.2-2 and 2.2-3 are modified to accommodate the implementation of the ESCU methodology used originally in the thermal-hydraulic analysis of the reference cycle (Unit 1 Cycle 10) and now in the Unit 2 Cycle 9 reload design. Use of the ESCU methodology requires changes to the coefficients of the PVAR equation presented in Figures 2.2-2 and 2.2-3, ensuring agreement of the Thermal Margin/Low Pressure trip setpoint with the Unit 2 Cycle 9 DNBR analysis. Therefore, these changes are acceptable.

b. Axial Power Distribution Trip Loss

TS Figure 2.2-1, "Peripheral Axial Shape Index, Y_I vs Fraction of Rated Thermal Power," is modified by increasing the acceptable operation region below 70% Rated Thermal Power and by shaping the negative Axial Shape Index (ASI) side to accommodate the increased core average linear heat generation rate (CALHGR) of Unit 2 Cycle 9 which is the second 24-month cycle in Unit 2. The CALHGR is increased for Cycle 9 because of the increased number of B_4C shims. These modifications were considered in the Unit 2 Cycle 9 reload safety analyses as well as in the setpoint analysis and acceptable results were obtained. Consequently, these changes are acceptable.

TS Basis B 2.2.1 is modified to identify the DNBR Specified Acceptable Fuel Design Limit (SAFDL) value of 1.15. This change is resultant from use of the ESCU methodology and is identical to that approved for the reference cycle and is acceptable.

c. Shutdown Margin

TS 3.1.1.1 and 4.1.1.1.1 (with the addition of new Figure 3.1-1b) are modified to establish a shutdown margin limit for Unit 2 Cycle 9 as a function of time in cycle. The actual required shutdown margin varies throughout the cycle due to changes in the core such as fuel depletion, boron concentration, and moderator temperature. The TS references the new Figure 3.1-1b, which provides a shutdown margin limit line varying from 3.5% delta rho required shutdown margin at BOC to 5.0% delta rho shutdown margin at EOC. The Unit 2 Cycle 9 safety analyses consider the limits of Figure 3.1-1b and are bounded by the reference cycle. This change is identical to the shutdown margin change approved for the reference cycle and as so, is acceptable.

TS Bases B 3/4.1.1.1 and B 3/4.1.1.2 are modified to explain the basis of the proposed change in the shutdown margin limit. This change is acceptable.

d. CEA Group Insertion Limits

TS Figure 3.1-2, "CEA Group Insertion Limits vs Fraction of Allowable Thermal Power for Existing RCP Combination," is modified to allow greater insertion of Group 5 CEA's between 90% and 100% power. Specifically, Group 5 insertion is changed from 25% at 100% power to 35% insertion between 90% and 100% power. This change allows an additional 13.5 inches

of CEA insertion of the lead regulating group for enhanced control of axial power oscillations. The Unit 2 Cycle 9 reload safety analyses assume the proposed limits of Figure 3.1-2. This change is identical to the Figure 3.1-2 change approved in the reference cycle and is acceptable.

e. Peaking Factors

TS Limiting Condition for Operation (LCO) 3.1.3.1 and Figure 3.1-3, "Allowable Time to Realign CEA Versus Initial Total Integrated Radial Peaking Factor," are modified to incorporate an increase in the maximum allowed F_r^I from 1.65 to 1.70. This is done to accommodate the increased neutron flux peaking associated with the second 24-month cycle for Unit 2. The setpoint analysis performed in the support of Unit 2 Cycle 9 considers this proposed change in F_r^I. Also, this TS reflects the use of the CECOR 3.3/BASSS network which is considered in the Unit 2 Cycle 9 setpoint analysis. Therefore, the changes are acceptable. The text of TS Basis B3/4.1.3 is modified to reflect these changes.

Figure 3.2-3b, "Total Planar Radial Peaking Factor vs N," is modified to indicate a reduction in its acceptable value region due to a reduction in the 100% power F_{xy}^I value of 1.54 to 1.50. This reduction is needed to accommodate the increased core average linear heat generation rate of the second Unit 2 24-month reload core design with its higher number of B_4C shims. This change is acceptable as the setpoint analysis takes credit for this modification in demonstrating acceptable results for Cycle 9.

Implementation of CECOR 3.3/BASSS as the on-line incore LCO monitoring system necessitated the proposed changes in TS 3.2.2.1, 3.2.3, 4.2.2.1.2, 4.2.2.1.3, 4.2.2.1.4, 4.2.2.2.2, 4.2.2.2.3, 4.2.2.2.4, 4.2.3.2, 4.2.3.3, 4.2.3.4, 4.2.5.3 and B 3/4.1.3 to ensure they adequately consider the CECOR 3.3/BASSS system. This implementation was shown to be acceptable in Section 3.5 of this SER.

LCO 3.2.3 and TS Figure 3.2-3c, "Total Integrated Radial Peaking Factor vs Allowable Fraction of Rated Thermal Power" are modified to reflect the increase in F_r^I from 1.65 to 1.70 to accommodate the increased nuclear flux peaking associated with this second 24-month cycle for Unit 2 and implementation of the CECOR 3.3/BASS on-line incore monitoring system. The Unit 2 Cycle 9 setpoint analyses, which was performed using NRC approved methodology, supports this change in F_r^I and, hence, the change is acceptable.

f. Peak Linear Heat Rate

Figure 3.2-1, "Allowable Peak Linear Heat Rate vs Burnup," is modified to indicate the reduced allowable peak linear heat rate (APLHR), for Unit 2 Cycle 9, from 15.5 Kw/ft to 15.2 Kw/ft. The APLHR is reduced to maintain the maximum Cycle 9 internal pin pressure below reactor coolant system pressure of 2250 psia. This change in APLHR is considered in the Unit 2 Cycle 9 fuel performance analysis. Also, the LOCA analyses have shown that a PLHR as high as 15.5 Kw/ft complies with the acceptance criteria. Therefore, this change is acceptable.

g. DNB Parameters

The text of 3.2.5 and Table 3.2-1 is modified by changing the phrase "core power" to "thermal power" to maintain consistency in the TS. This change is administrative in nature and is therefore acceptable.

TS Basis B 3/4.2.5 is modified to specify the new DNBR SAFDL value if 1.15 which results from use of the ESCU methodology. The change is acceptable.

h. Auxiliary Feedwater Flow Rate

TS Basis B 3/4.7.1.2 is modified by increasing the maximum allowed auxiliary feedwater flow from 1300 gpm to 1550 gpm. An evaluation of the effects of increasing this flow was performed by the licensee and it was determined that the results on the safety analyses for Unit 2 Cycle 9 are bounded by previously reported and approved analyses. The change is acceptable.

i. Axial Shape Index

TS Figure 3.2-2, "Linear Heat Rate Axial Flux Offset Control Limits" and Figure 3.2-4, "DNB Axial Flux Offset Control Limits," are modified to increase the negative ASI limits below 50% power. The licensee has evaluated the effect of the proposed new limits on the Unit 2 Cycle 9 transient analyses, margin to fuel centerline melt limits, margin to DNB limits, margin to LOCA PLHGR limit, core power versus planar radial peaking factor LCO, TM/LP LSSS, and core power versus integrated radial peaking factor LCO and has determined that acceptable results are obtained. Thus, these changes are acceptable.

j. Core Enrichment

TS 5.3.1 is modified to indicate an increase in the maximum enrichment for a reload core from 4.1 weight percent (w/o) to 4.35 w/o U-235. This change is proposed to allow the higher enriched Unit 2 Cycle 9 second 24-month cycle reload core. All aspects of the Cycle 9 reload core design consider the proposed higher enrichment. The Unit 1 and 2 fuel storage facilities have been approved for storage of fuel of maximum enrichment of 5.0 w/o U-235. This approach was provided in the NRC Safety Evaluation dated January 10, 1990. Therefore, this change is acceptable.

7.0 STARTUP TEST PROGRAM

The startup testing program proposed for Unit 2 Cycle 9 is similar to that used in the reference cycle. However, a change is proposed in the manner in which core symmetry is verified by using incore power distribution monitoring with only minimal CEA symmetry testing. This monitoring would be done using measured power distributions generated by the new CECOR 3.3 system which does the solution in full core as contrasted to the previous INCA system whose solution is for an eighth core. The staff has approved this new monitoring system in Section 3.5 of this SER. The licensee has also performed an

analysis using approved methodologies which demonstrates that the incore monitoring in full core, in conjunction with minimal CEA symmetry testing, will detect any significant fuel assembly misloadings. Therefore, the staff finds this proposed change in core symmetry confirmation to be acceptable.

8.0 SUMMARY

The staff has reviewed the fuel system design, nuclear design, thermal-hydraulic design, startup test program, and the transient and accident analysis information presented in the Calvert Cliffs Unit 2 Cycle 9 reload submittals. In addition, the modifications made to the refueling procedures as a result of the licensee's 10 CFR 21 notification concerning potential loss of shutdown margin were also reviewed.

Based on this review, which is described above, the staff concludes that the proposed Unit 2 Cycle 9 reload and associated modified TS are acceptable. This conclusion is further based on the following: (1) previously reviewed and approved methods were used in the analyses; (2) the results of the safety analyses show that all safety criteria are met; and (3) the proposed TS are consistent with the reload safety analyses.

9.0 INTENT

The intent of the proposed changes is to authorize Unit 2 Cycle 9 operation.

10.0 STATE CONSULTATIONS

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the Federal Register on April 12, 1989 (54 FR 14714). No hearing requests or intervention petitions were received. The State of Maryland was consulted on this matter and had no comments on the determination.

11.0 ENVIRONMENTAL CONSIDERATION

This amendment, with the exception of changes in the enrichment limits for fuel located in the Unit 2 core, involves a change in the installation or use of the facility's components located within the restricted areas as defined in 10 CFR 20 and changes in surveillance requirement. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards considerations and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

As for the changes in TS core enrichment limits, pursuant to 10 CFR 5.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on January 30, 1989 (53 FR 4352). Based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect upon the quality of the human environment.

12.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 10, 1990

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PACKAGE DIVIDER

DRP-003